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TEST REACTOR RISK ASSESSMENT METHODOLOGY

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ABSTRACT

A methodology has been developed for the identification of accident initiating events and the fault modeling of systems, including common mode identification, as these methods are applied in overall test reactor risk assessment. The methods are exemplified by a determination of risks related to a loss of primary coolant flow in the Engineering Test Reactor.

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TEST REACTOR RISK ASSESSMENT METHODOLOGY

I. INTRODUCTION

The methods developed and described herein are intended for use in quantitatively evaluating the risks of test reactors. Risk is generally defined in the dictionary as the chance of injury where injury might be construed to mean bodily injury, loss of wages, loss of property, etc. In terms of accident producing injuries risk can be defined as the product of an accident occurrence probability and the consequence of that accident (that is, risk equals accident probability times accident consequence). In reactor safety it has been customary to talk about the probability of accident occurrence in a qualitative sense. For example, the familiar terms credible, incredible, likely, unlikely, extremely unlikely, and anticipated are usually used to describe the predicted frequency of accidents. The consequence factor, on the other hand, has been subject to the rigor of the scientific approach and, therefore, can be expressed quantitatively. Unless both factors in the risk expression are quantified, risk and, therefore, safety must be generally classified as unknown.

The recent Reactor Safety Study (RSS)[1] was a comprehensive study to estimate quantitatively the risks associated with two large commercial nuclear plants and to compare those risks with other natural and man-made risks to which the general public is exposed. The methods used to assess risks by the RSS can be used to evaluate any type of reactor plant. However, areas exist in which modification and expansion of the methods used by the RSS is desirable so that the methods can be made more suitable to the evaluation of other plants. The primary objective of the present study is to develop methods which can be applied to ERDA-owned test reactors, the intent being to modify and expand the methodology used in the RSS where needed and to improve the methodology where improvement is clearly possible. A secondary objective of this study is to demonstrate the methods which are developed by applying them, on a skeleton basis, to the evaluation of the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory (INEL).

Three major areas exist in which departures have been made herein from the methods used in the RSS: (a) accident initiating event identification, (b) system fault model construction, including the treatment of common mode events, and (c) radiological consequence determination. Accident initiating event identification and system fault modeling have been subject to development by this study. They are discussed in some detail in Sections III and IV, respectively. The discussion on fault modeling duplicates somewhat that contained in Reference 1 because it was believed to be needed in order to facilitate description of the improved method. Radiological consequence determination was not subject to methodology development. Standard methods are in use at the INEL for making these determinations and were used herein to illustrate the risk methods. This alternate methodology for determining radiological consequences is discussed in Section II. Also in Section II is an overview of the tasks required to do a risk evaluation.

A few terms are used herein that required definition. These terms are defined in Section V.

A hypothetical loss of flow accident at the ETR was selected to illustrate the risk assessment techniques. This sample assessment is described in Appendix A.

II. RISK ASSESSMENT

The major tasks proposed here for assessing test reactor risks are summarized in the following discussion. The overall approach is generally the same as that used in the RSS which is shown in Figure 1. Where departures have been made from that study, they are so mentioned. For more detailed discussion regarding the methodology used in that study, the reader should refer thereto.

1. ACCIDENT INITIATING EVENTS

The first step in a risk assessment is to identify the accident initiating events associated with the plant; that is, to identify those events such as loss of coolant or loss of flow that could result in a radioactivity release[a] if the safety systems provided to cope with those events did not respond. The accident initiating events selected in the RSS were those recognized by the Nuclear Regulatory Commission as important to nuclear safety and events which have evolved over the years from reactor licensing activities and extensive reactor safety research. The test reactor designs have not been subject to that same research. Instead, they are precursors to and specifically designed tools used in that research. Major differences occur in the objectives of the two reactor types and, therefore, their designs; so the accidents, and risks, could appreciably differ[b]. And faults occurring in experiments, which might change from test to test, could change the nature of the risks from experiment to experiment. Appropriately then, a procedure should be developed which could be used to systematically identify accident initiating events. Such a procedure has been developed here; it makes use of the fault tree method to (a) identify all sources of radioactivity within a test reactor facility, (b) identify the barriers which retain that radioactivity, and (c) identify the mechanisms by which those barriers can fail. The failure mechanisms are the accident initiating events. This procedure is discussed fully in Section III.

2. EVENT TREES

The second step in the procedure is to develop an event tree for each of the important accident initiating events identified. The event tree is a logic diagram which depicts all meaningful outcomes, or accident sequences, which result from each applicable engineered safety feature, either responding or not responding to a given accident initiating event. For example, typical possible responses to a large pipe break in a power reactor might be those shown in Figure 2. Each of the systems needed to respond to a pipe break is shown

[a] The scope of this study has been arbitrarily limited to those risks pertaining to radioactivity releases.

[b] Some of the major differences between test reactors and large commercial reactors that could appreciably affect their risk differences are discussed in Appendix B.

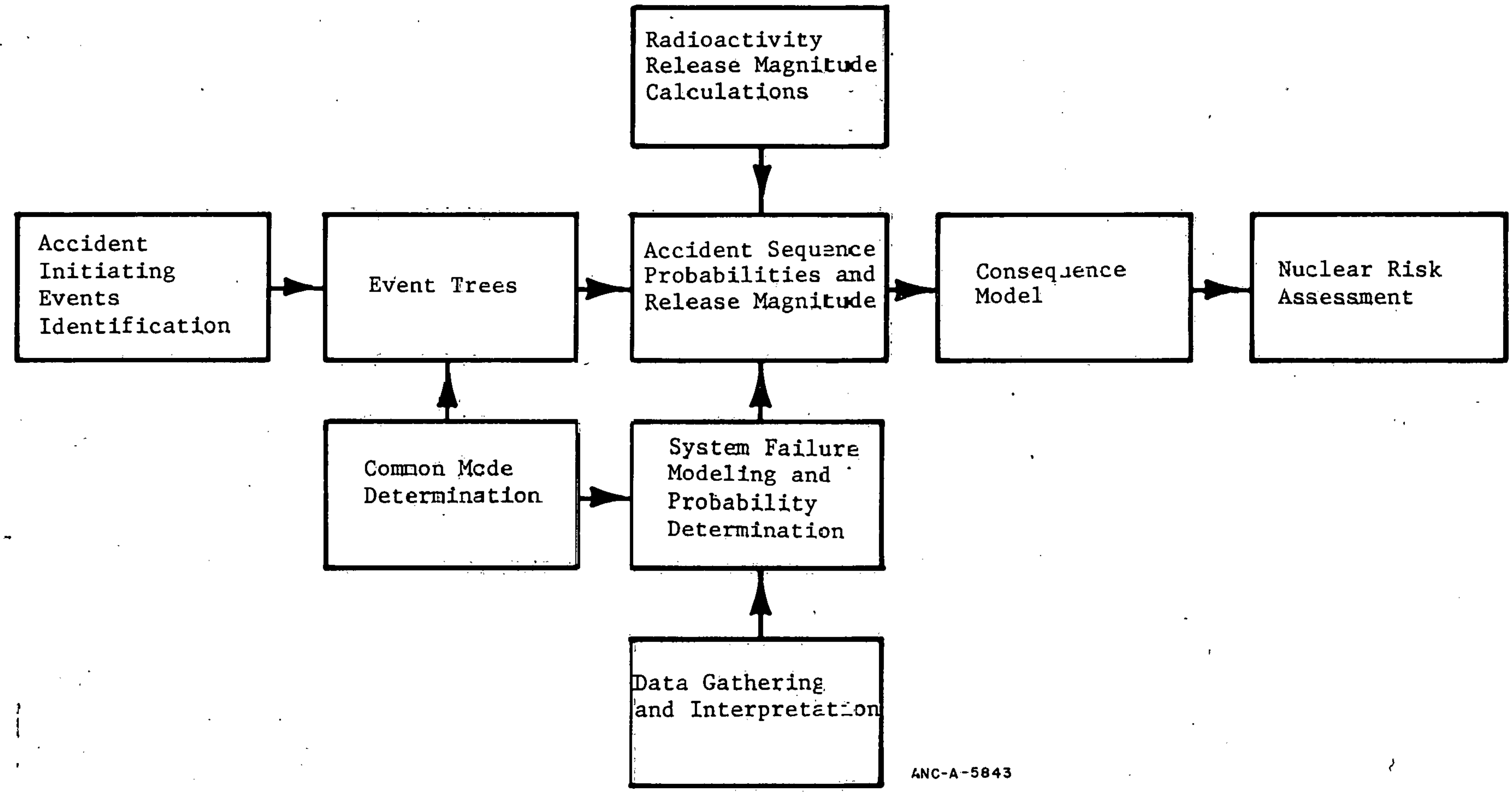

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Fig. 1 Risk assessment tasks.

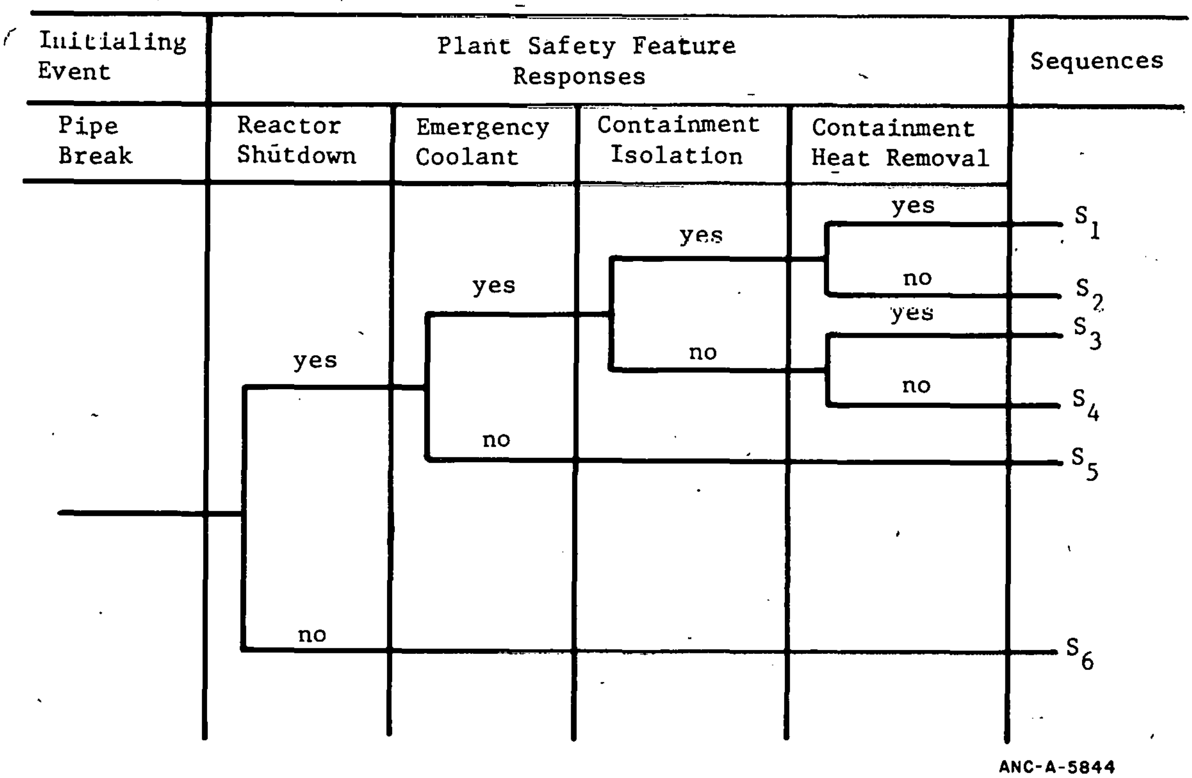


Fig. 2 Example event tree diagram.

as a column heading on the event tree, and each of these systems can succeed (denoted by yes in the example) or fail (denoted by no). Whether a system works or does not work results in a different sequence. The column headings are ordered according to time and dependency such that the system to the left must work first, those to the right last; dependent systems are shown first, nondependent systems are shown last. Logic reasoning is applied in the construction of the event tree so that meaningless sequences are eliminated. For example, if the reactor shutdown system does not work, whether or not the emergency coolant, containment isolation, and containment heat removal systems operate is immaterial. Thus, a straight line is drawn across columns subsequent to the reactor shutdown failure.

The event tree, in addition to providing a logic diagram of the meaningful responses to an accident initiating event, serves to identify the systems requiring detailed analyses and it can be used conveniently to summarize the probability and consequence values obtained from the study. These values are discussed further.

3. SYSTEM FAILURE AND EVENT TREE SEQUENCE PROBABILITIES

Each sequence shown on the event tree has a probability of occurring and results in a definable radioactivity release. Radioactivity releases and their attendant consequences are discussed in Section II-4. Sequence probabilities are obtained by combining accident initiating event occurrence probabilities with probability values for each of the engineered safety systems identified in the individual sequences. Dependencies between systems identified in the sequences must be considered in the computations, and can be done automatically for cases in which the complete systems fault models can be combined before the computation is made.

The probability values used in a sequence computation are the probability that the initiating event occurs and the probability that each of the responding systems is in a failed state (does not work as shown in Figure 1) when the initiating event occurs (system unavailability).

System failure probabilities are derived by developing detailed fault models of each of the systems which appear in the accident sequences, applying experience data to those models, and combining those data according to the logic of the models. Tasks 5, 6, and 7 of the RSS shown in Figure 1 embody these activities. The RSS found the fault modeling of systems to be very time consuming and, therefore,costly. So considerable effort was devoted to streamlining the fault tree methodology, and as a result an abbreviated approach has been developed and tried which reduces by about one-half the time required to develop a conventional fault tree model. This modified procedure has other advantages over the conventional fault tree approach in that it provides more visibility of the system fault modes and their logical interrelationships, modifications are made relatively easy, and the model can be checked against the system without much difficulty. The procedure also describes a workable approach for the identification of common mode failures which are readily incorporated into the fault model. This abbreviated fault modeling procedure is described in Section IV.

4. CONSEQUENCES

As indicated in Section II-3, each accident sequence results in a definable radioactivity release and ultimately results in a definable consequence. The RSS, in essence, determined the release for each sequence (Task 3 in Figure 1). Actually, a large number of sequences involved similar physical processes and produced similar consequences; thus, the RSS was able to define a few release categories in which the sequences could be grouped. The result was a histogram representing the probability distribution of the release categories. This histogram was then input to the consequence model. The consequence model, in essence, then combined each of the values in the histogram with each of the values in distributions for other important variables until all possible combinations of the values were considered (Task 8, Figure 1). The result of the calculation was a probability versus magnitude distribution for each of the consequences: deaths, injuries, long-term health effects, and property damage. These results were presented in graphical form with the probability of a consequence equal to or greater than X plotted along the ordinate and consequence X plotted along the abscissa (Task 9 in Figure 1).

A different approach is used here for dealing with the consequence factor in the risk expression. Primarily, the release probabilities are not input to the consequence model. Instead, the consequence of each sequence is determined independently of the sequence probability, and the result is multiplied by the sequence probability to obtain a risk value for that sequence. The probability, consequence, and risk value corresponding to each sequence is then summarized on the event tree. (The event tree, Figure A-12, in Appendix A reflects these values for a loss of flow event at the ETR.)

The radioactivity release terms were obtained by the RSS through use of a computer code called CORRAL developed in that study. Here a computer code, RELAP4[2], is used to calculate the fraction of core damage and this calculation, in turn, provides input to a computer code, RSAC2[3], which calculates the consequences.

RELAP4 is a thermal analysis code developed by Aerojet Nuclear Company which describes the behavior of water-cooled reactors subjected to postulated transients such as loss of coolant, pump failure, or excursions in nuclear power. Inputs to the code describe the geometry of the system and initial conditions; the code calculates as a function of time fluid conditions (for example, flow, pressure, and quality), thermal conditions (for example, surface temperatures and temperature profiles), heat fluxes in power generating and dissipating elements, reactor power, decay heat, and reactivity.

RSAC2 calculates radiological consequences resulting from reactor accidents. Input to the computer code generally consists of the reactor core power history, fission product release data, and data which describe the meteorology of the site (for example, release height, inversion height, meteorological diffusion data, wind speed, release rate, and downwind distance). With fission product dispersion calculated, radiological doses for hypothetical receptors at predetermined downwind locations are calculated. These radiological consequences can be expressed in terms of wholebody gamma doses due to immersion in the radioactive plume or in terms of body organ doses by specific nuclides due to inhalation.

III. ACCIDENT INITIATING EVENT IDENTIFICATION

The first step in risk assessment is to select the accident initiating events, that is, those events that can result in accidents if not checked or mitigated. For this study, the events that could lead to release of radioactivity must be found. These events may be operating errors or hardware failures that are not expected to occur but, should they occur, may require operator or safety system action to prevent the release of radioactivity. The potential of these events occurring combined with the potential that the corrective action is not taken is part of the risk calculation. The consideration of all accident initiating events is essential for an accurate assessment of risk.

The steps for identifying accident initiating events are shown in the block diagram in Figure 3. As indicated in the left block, the first step is to state the undesired event, in this case a radiological release from the reactor facility. This is the release for which the probability of occurrence and consequence will be determined.

The second step is to determine all possible sources of radioactive material at the reactor facility that if released could cause the release stated in Step 1.

Step 3 requires consideration of the plant operating states that can affect the release magnitude. For example, the release potential will be different if flow blockage occurs in the reactor cooling system when the reactor is or is not operating.

The barriers that prevent the release of the radioactive material from each potential source are determined in Step 4.

The final step lists the accident initiating events. These events are the mechanisms that can cause the failure of each barrier. Event trees are developed for each of the events identified in this step as part of the overall risk assessment.

The fault tree method illustrated in Figure 4 is used when Steps 1 through 5 are performed in order to logically tie together the steps which lead to the listing of all accident initiating events. The statement of the undesired radiological event becomes the top event for the fault tree. The analyst is then required to determine all possible sources of radioactive material within the facility that if released could cause the top event to occur. Information must be obtained about the flow of radioactive material at the reactor site for the analyst to be able to make this list.

The next level of the tree contains the "house" fault tree symbol that is used here as a switch to "turn on" that part of the tree that should be included and thus allows investigation of radiological releases of different magnitudes. The analysis can be general or explicit as to the amount of release; here the amount is kept general because the amount does not enter into the calculations. This switching ability of the "house" is explained in Section IV-7.

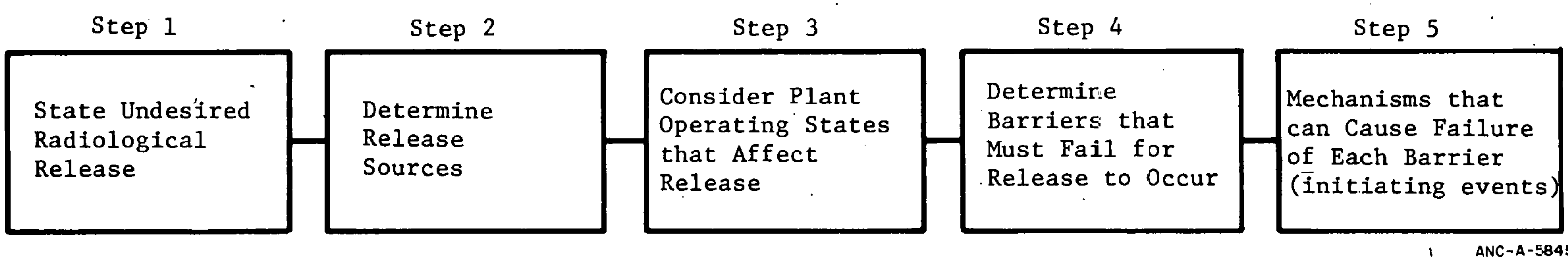


Fig. 3 Major steps to identify accident initiating events.

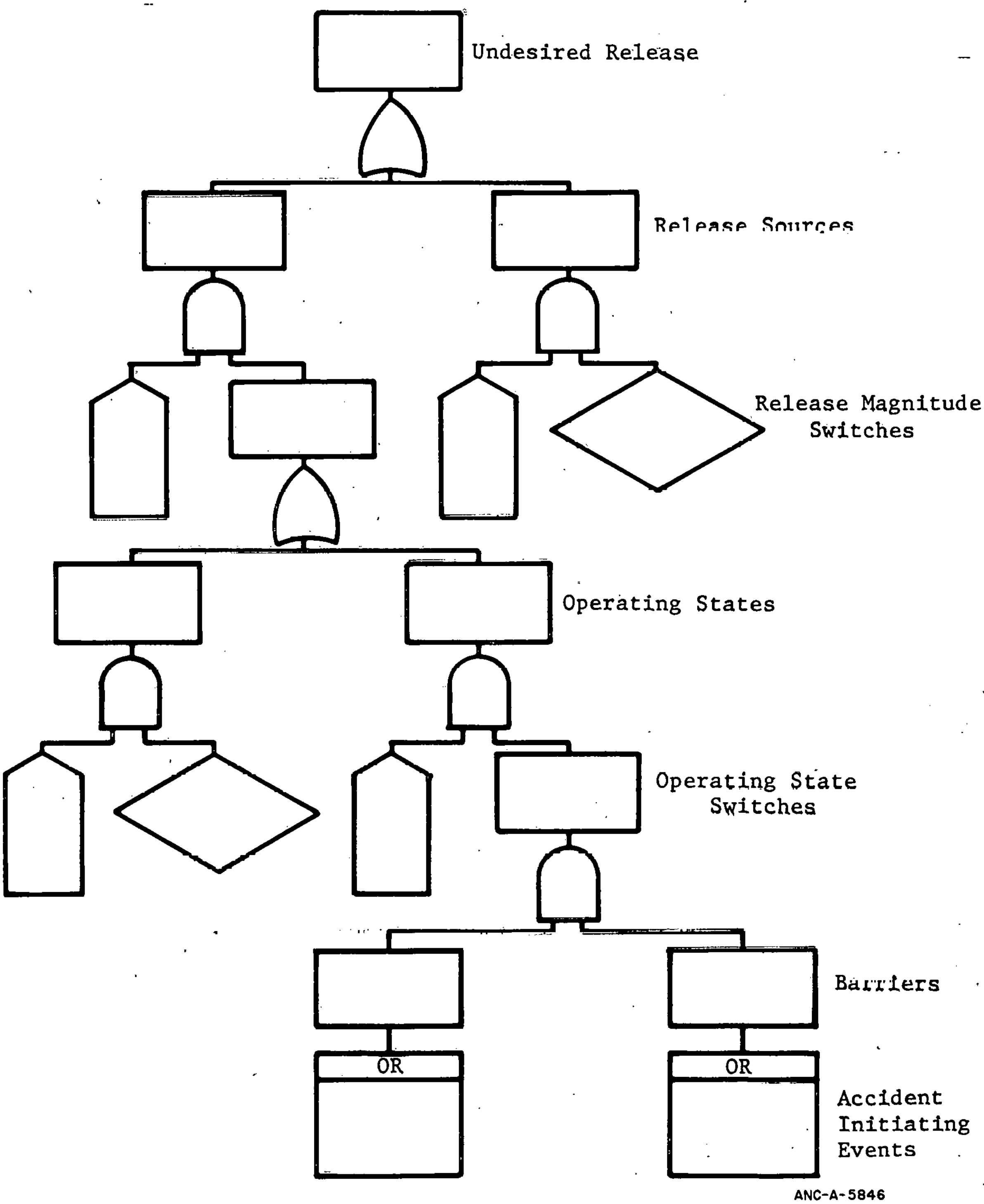


Fig. 4 Fault tree logic leading to list of accident initiating events.

If different operating states affect the magnitude of the release from a source, the next level of the tree should distinguish these operating states by using "houses" at the next fault tree level to switch on or off operating states to be considered or not considered, respectively, in the analysis.

Before the radioactive material at each source can be released to the environment, the barriers that contain the material must be penetrated. Each barrier becomes an input to an "AND gate" on the fault tree. All the barriers for each source must be penetrated before the release can occur.

The accident initiating events are those events that can cause failure of the barriers. All mechanisms of failure should be considered. The list should be verified by knowledgeable individuals and by a study of the reactor operating history.

IV. SYSTEM FAULT MODELING

A modified and abbreviated version of the fault tree method is used to determine system success and failure probabilities where the system, in turn, is related to the overall risks associated with the plant. Fault tree analysis is a systematic procedure used to identify and record the various combinations of component fault states that can result in a predefined undesired state of a system. Unlike the familiar inductive method of first postulating a component failure mode and then determining its effect on the system, fault tree analysis is an opposite deductive approach whereby the analyst first defines an undesired system effect and then identifies all the component failure modes that can, by themselves or in combination with other component failure modes, produce that predefined undesired system state. A fault tree, as opposed to fault tree analysis, is a result of the fault tree analysis and is a graphic display of all of the fault modes of a system and the combinatorial AND and OR logic that relates those fault modes to the predefined undesired event associated with the system. It is a fault model of the system which, when expressed in its nonredundant Boolean form, can be used as a probabilistic model to determine an overall probability of system failure based on known, or easily computed, probabilitiy values for individual events shown on the fault tree.

1. SYSTEM FAILURE DEFINITION AND UNDESIRED EVENT

Fault tree analysis begins with a statement of the undesired event. Embodied in that statement must be the conditions which constitute failure of the system. For example, the undesired event, "insufficient coolant flow through the reactor core when the reactor is generating heat" is considered. This event statement is a complete logic statement specifying the requirements for reactor coolant. If a fault tree were to be developed about the undesired event, the analyst would examine all systems, normal operating and emergency systems, which deliver coolant to the reactor vessel. The analyst may define a more restrictive undesired event, for example, "insufficient emergency coolant flow when normal flow is lost", for which a fault tree is developed for the auxiliary coolant systems only.

The undesired event examples previously presented are stated rather generally which in most cases is perfectly acceptable. For example, the word "insufficient" implies that below some flow value, the system will have failed. Where redundancy has been provided, however, the generalized statement must be translated into a statement more specific in order to account for the redundant capabilities of the system. For example, the statement "insufficient coolant flow . . ." might be translated into the more specific statement "less than two pump coolant flow . . ." where more than two pumps have been provided.

The fault tree will be developed about the selected undesired event, and only events which relate logically to the occurrence of that undesired event will be identified.

Component failures that produce other undesired events (for example, inadvertent operation of the system) when loss of flow is of concern will not be identified unless the particular component failures relate to the occurrence of both undesired events.

The undesired event and all subsequent events shown on the fault tree are binary. That is, if the event as stated occurs, the system (or component in more detailed parts of the tree) has failed; if the event does not occur, the system has not failed. Ambiguous or "maybe" statements are not allowed on the tree. The statement is either true if the condition exists or false if the condition does not exist.

2. FAULT TREE CONSTRUCTION

Once an undesired event has been defined, a fault tree can be constructed about that undesired event. To illustrate the procedure, the ETR core deluge system will be used as an example. First, the top tiers of the fault tree will be constructed using the conventional method[4,5]; then the tree will be restructured using an abbreviated approach developed and demonstrated herein.

Figure 5 is a simplified schematic of the core deluge system. The system is comprised of two redundant subsystems, each containing an air-operated block valve which must open when the system is needed to allow emergency coolant to flow from a source of coolant to the reactor vessel. The block valve is opened automatically by associated instrumentation and controls when the water in the reactor vessel drops below a predetermined safe level.

2.1 Conventional Fault Tree Construction

The top event selected for the core deluge system is "insufficient coolant flow from the core deluge system when the reactor is producing heat and the water level in the vessel is low". This event, as shown in Figure 6, is the top event on the fault tree. Since the "when" statement specifies the conditions under which the fault events to be defined by the fault tree produce system failure (Section IV-7), the top undesired event is translated into two statements: (a) "reactor producing heat and vessel water level low", which is shown within a house, and (b) "insufficient coolant flow from the core deluge system", a fault, shown within a rectangle, which will be developed further. Since there are two subsystems, either one of which will provide sufficient emergency coolant, the "insufficient coolant flow from the core deluge system" statement can be translated into two statements, "insufficient coolant from core deluge Subsystem A" and " insufficient coolant from core deluge Subsystem B". Both subsystems must coexist in a failed state in order to produce failure of the system; so an AND gate is used to relate the two subsystem events to the system event (gate types are discussed in Section IV-4). Thus a system event, a set, has been transformed into subsystem events, or subsets.

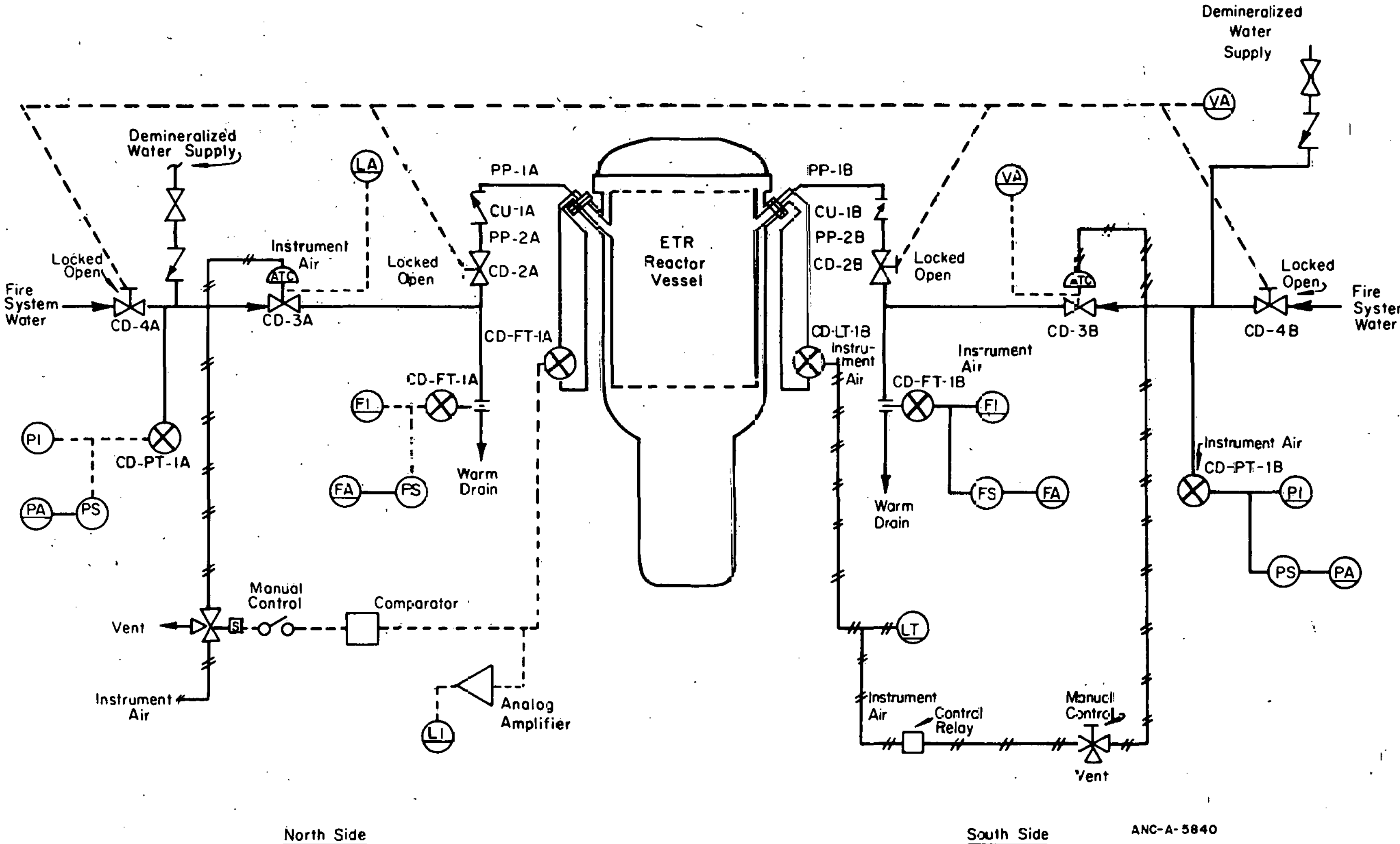


Fig. 5 Core deluge system simplified diagram.

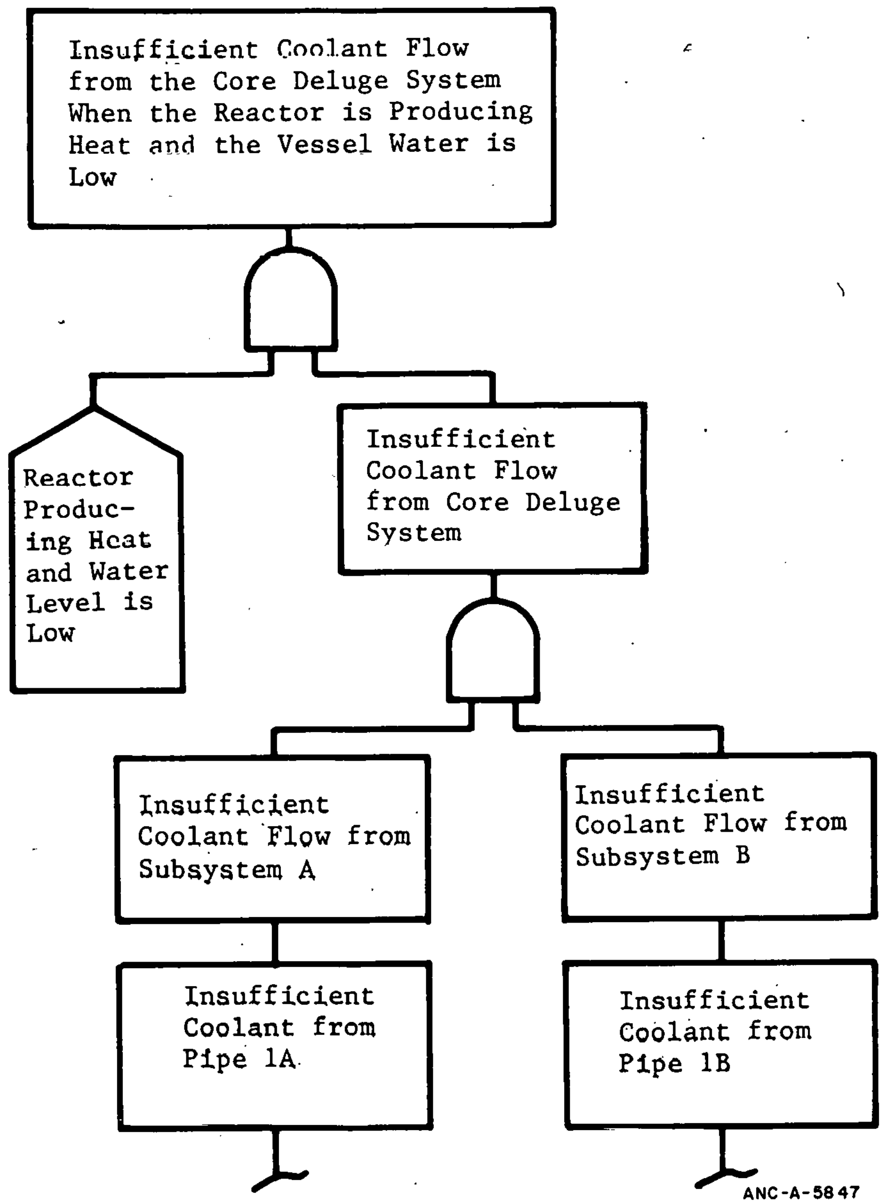


Fig. 6 Top level fault tree core deluge system.

Next, the subsystem events are translated into events pertaining to components within the subsystem. At this level of resolution, each component fault in a subsystem is enumerated in the subset, but no new levels are established. To enumerate component faults in an orderly fashion is desirable, and is most easily accomplished by starting with the subsystem output components and then stepwise examining each upstream component. That is, to work toward sources of energy is desirable because each succeeding tier of events on the fault tree results in the preceding or higher tier of events. So the subsystem event "insufficient coolant from Subsystem A" is translated into an event related to the output component of the subsystem, a pipe, which is "insufficient coolant flow from Pipe 1A". The Subsystem B event is similarly restated.

Examination of the fault tree in Figure 7 for output Pipe 1A shows that "insufficient flow from Pipe 1A" can be caused by rupture of Pipe 1A, plugging of Pipe 1A, a higher pressure in the reactor vessel than in the core deluge system, or insufficient coolant from the upstream-component Check Valve DV-1A. These are all the immediate events sufficient and necessary to prevent coolant flow through Pipe 1A.

Development of the fault tree one more step involves examination of the next upstream component of the subsystem to determine what can cause "insufficient coolant flow from Check Valve 1A". This additional development is shown in Figure 8. "Higher pressure in the vessel than in the system" would also be a cause for "insufficient coolant flow from Check Valve 1A"; however, this event has already been identified in the preceding component fault tree development; therefore, it is not restated (further discussion is provided in Section IV-8 on Boolean reduction for justification). Most component arrangements in typical systems are of the series type as indicated thus far – each component output is an input to a succeeding component. Some components have multiple inputs, and the inputs may relate to the output through AND logic as shown in Figure 9, OR logic as shown in Figure 10, or a more complex logic arrangement. Figure 11 is one example of complex inputs. A large number of possible complex logic arrangements occur, depending on the system and component complexity.

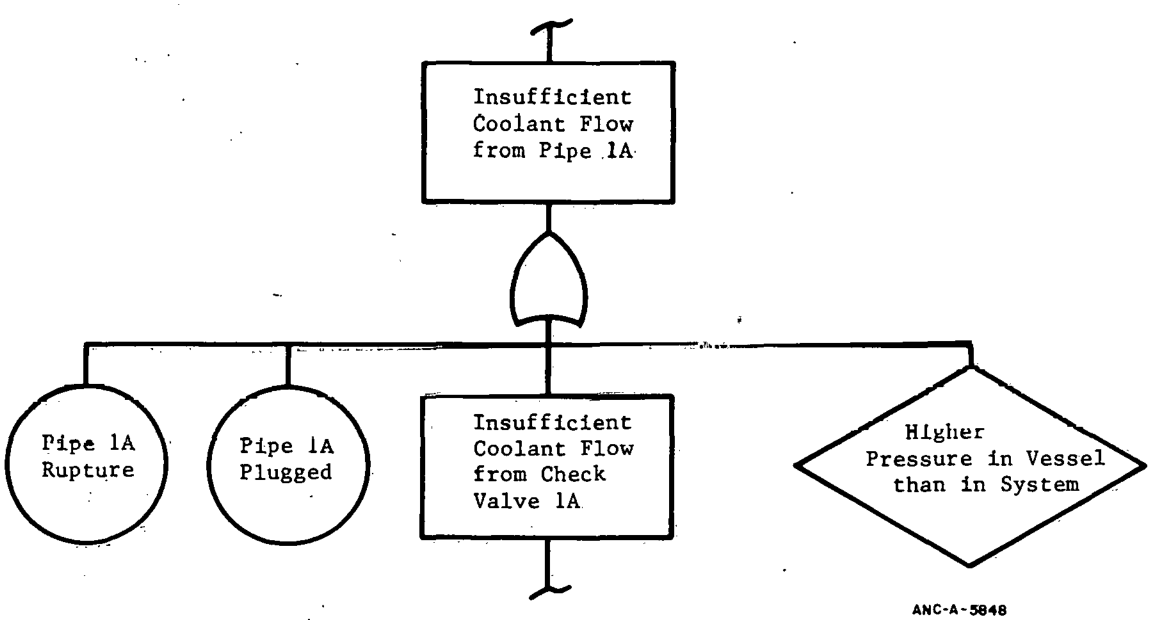


Fig. 7 Fault tree of Subsystem A output component.

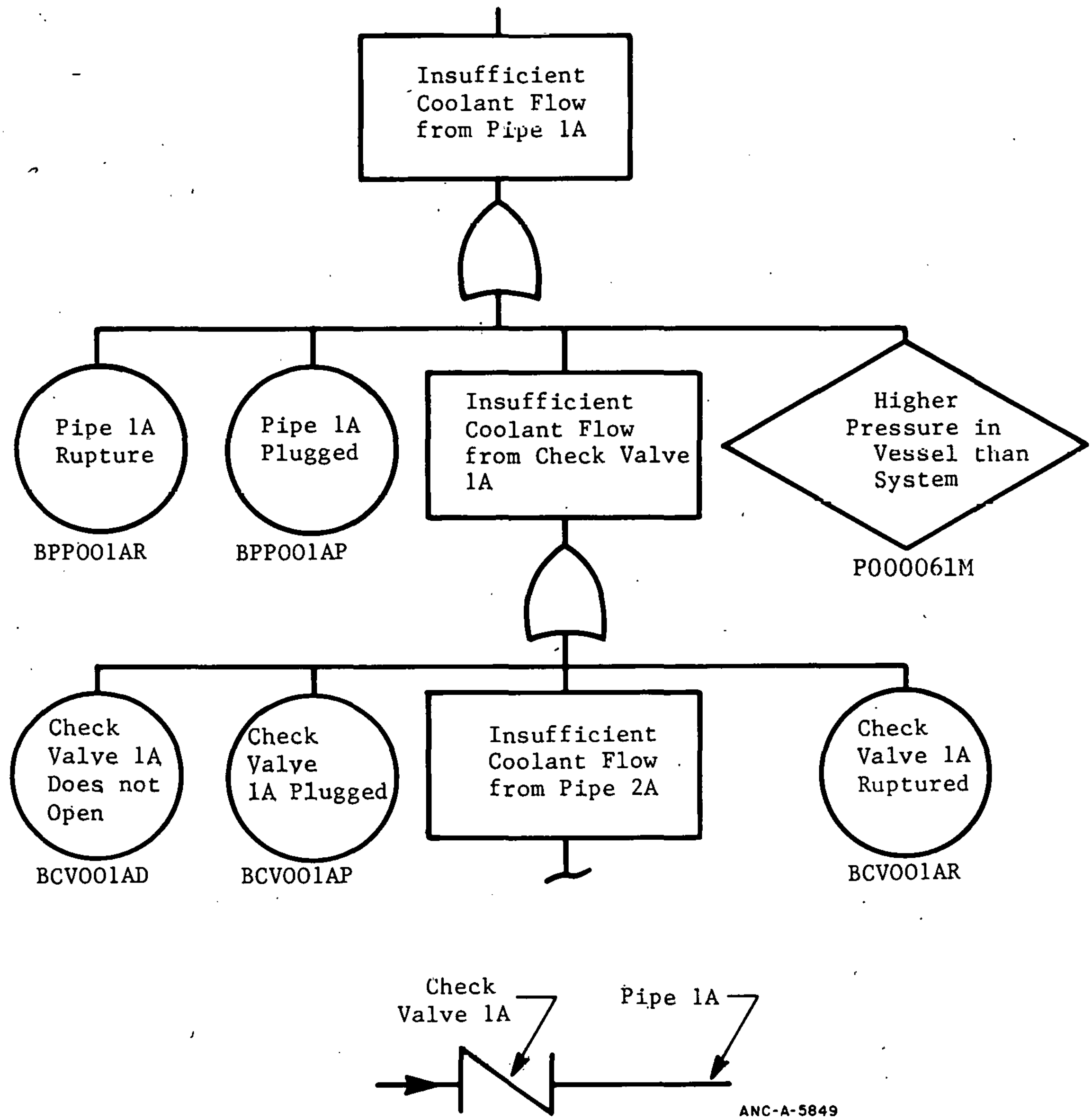


Fig. 8 Fault tree development of two single input components in series.

The fault tree is developed in the preceding manner until all components of the system are identified in their basic fault states. The result is a binary model of the system which can be reduced to its simplest Boolean form. Failure rates, human error rates, and appropriate time intervals can be assigned to determine probability values for the components, subsystems, and the system. The quantification process involves the naming of events (Event Naming, Section IV-6) and the transferring of all the information contained on the fault tree to event tables and coding sheets for ease in the assignment of data to events and for computer processing.

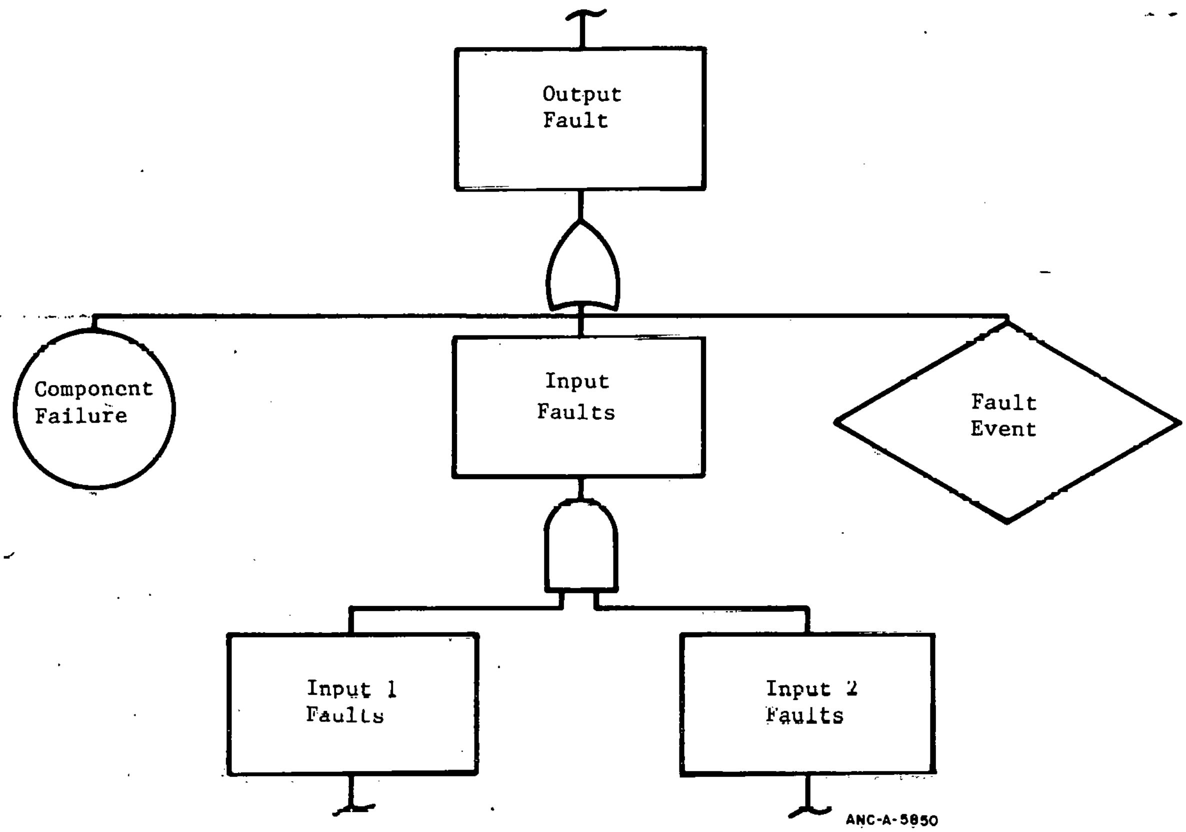


Fig. 9 Typical fault tree structure of component with two inputs related to output by AND logic.

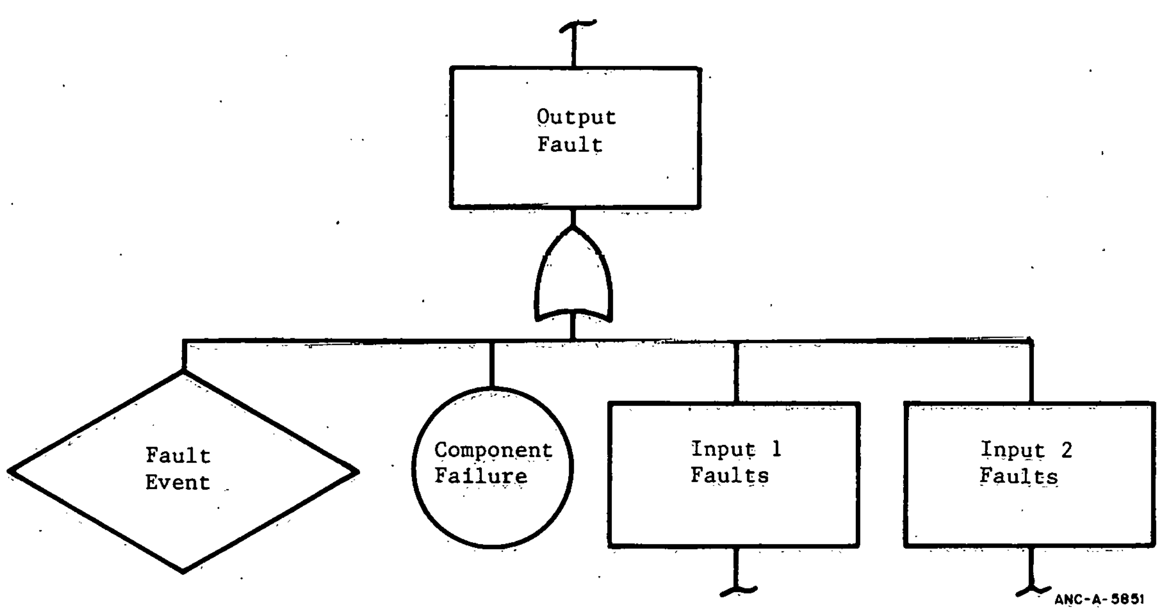


Fig. 10 Typical fault tree structure of component with two inputs related to output by OR logic.

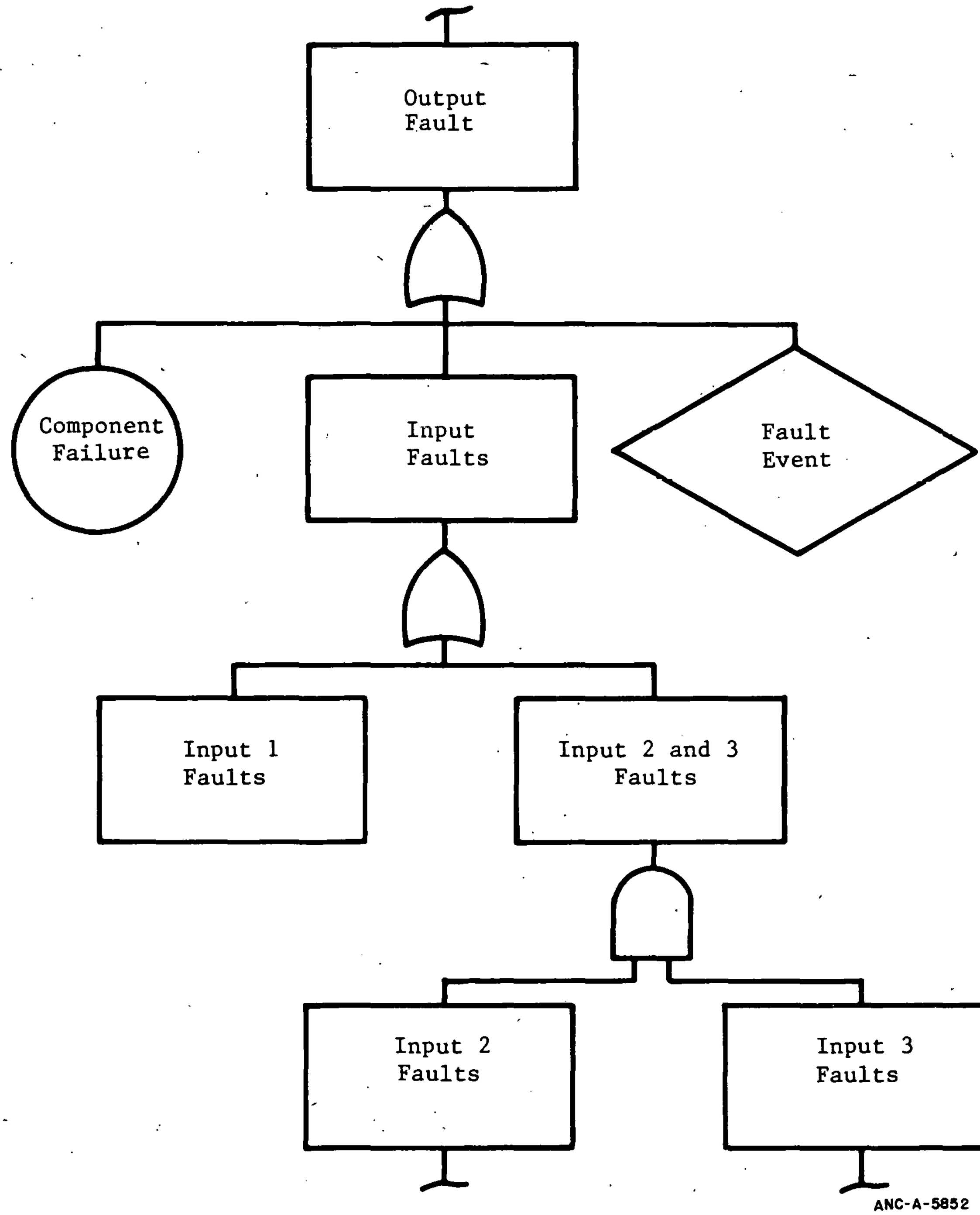


Fig. 11 Typical fault tree structure of component with complex inputs.

2.2 Abbreviated Fault Tree Construction

Since all basic fault event statements on the conventional fault tree are subsequently transferred to tables, one way to reduce the fault tree analysis effort is to not put those statements on the fault tree in the first place. The first step in the abbreviated method, then, is to enter all basic fault statements directly onto fault summary tables (a portion of a fault summary table is shown in Table I). Only the event code name, described in Section IV-6, is shown on the fault tree.

The second step in the procedure is to define a new logic gate, the tabulation OR gate (described in Section IV-4), to facilitate the listing of event names on the tree rather than to show named individual event statements within event type symbols as is conventionally done. Typically, systems which are evaluated contain a large number of events that are logically in series when reduced. For example, the fault tree development for the two core deluge components connected in series (shown in Figure 8) is considered. This development can be restructured as shown in Figure 12, where the code names for basic input events are listed under a tabulation OR gate, and only the input to the two components is identified for further development. The same treatment can be applied to any number of components logically in series. A completed fault tree for a system would be typically depicted by a top undesired event, basic fault events listed by code name under one or more tabulation OR gates, a few input events identified within rectangles which are inputs to chains of components and inputs to the system, a few house events, and the logic AND and OR gates used to relate the events. All the other information is contained in the fault summary table.

The abbreviated fault tree procedure has several distinct advantages over the conventional approach, all of which ultimately reduce the time and effort required to evaluate a system. Some of the more important of those advantages can be summarized as follows:

(1) Fault trees are readily restructured for each new accident situation. For example, the loss of flow accident associated with the ETR might result from any number of pipe ruptures or from failure of any number of faults in the power circuits to the primary coolant pumps. A loss of flow accident resulting from a pipe rupture may require that the core deluge system respond; whereas, a loss of flow accident caused by loss of power to the pumps requires that one of the emergency pumps continues to operate. A fault tree for each of the situations can be derived from a fault tree constructed for the general case of "loss of flow" by merely excluding on the fault tree all coded events related to loss of power to the primary pumps in the first case and excluding all pipe ruptures, also coded events, in the second case. The fault statements entered in the fault summary are not changed. During the course of this study the requirement for operation of primary coolant pumps changed from two of four pumps to three of four pumps. The change would have resulted in a major change to a conventional fault tree; only a few minutes were needed to change the abbreviated tree.

(2) Component fault modes and their logical relationship to system failure are more visible on the abbreviated fault tree. A typical system fault tree developed according to the conventional procedure usually requires 20 to 30 large sheets of paper in order to show all the component fault

TABLE I

FAULT SUMMARY

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^{-x})*									
			FAILURE RATE	FAULT DURATION	ERROF. FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HIGH TEMPERATURE	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP
BPPOO1AR	Pipe 1A	Rupture													
BPPOO1AP	Pipe 1A	Plugged													
POOOOO1M	Higher Pressure in Vessel Than System														
BCVOO1AD	Check Valve 1A	Does Not Open													
BCVOO1AP	Check Valve 1A	Plugged													
BCVOO1AR	Check Valve 1A	Rupture													

* PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

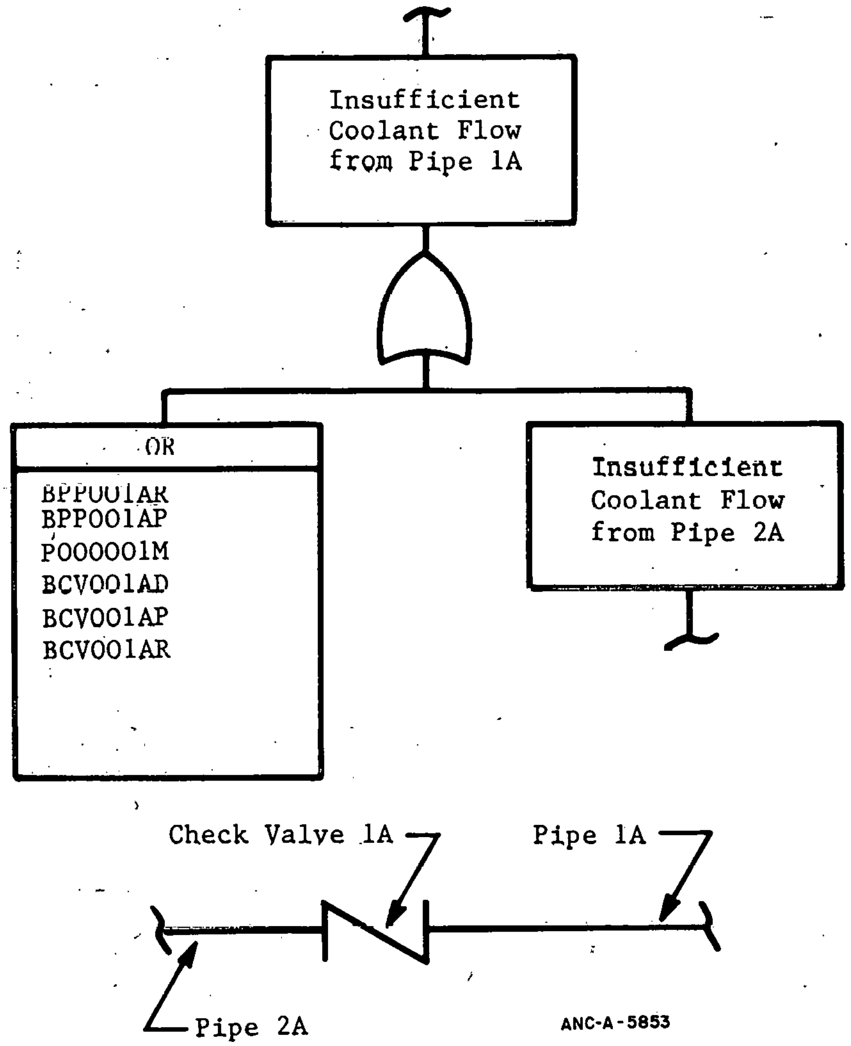


Fig. 12 Abbreviated fault tree structure of two components in series.

statements. These same component faults usually can be shown on two or three 8-1/2 x 11 inch sheets when presented in the abbreviated form. Because of their reduced size and because of the improved fault mode visibility of the system, the fault trees are much easier to check.

(3) A system evaluation is easier to stage using the abbreviated method. Analysis staging is discussed more fully in Section IV-12.

(4) The abbreviated procedure is more amenable to the treatment of common mode. This procedure is discussed in Section IV-9.

(5) Where formalized reports are required, most diagrams are superseded by tables which require less publication effort.

3. COMPONENT FAULT STATES

A component can transfer to a fault state due to any one of three categories of causes: primary failure, secondary failure, and command transition. A primary failure is the so-called "random" failure found in the reliability literature and refers to failure from no known external causes. A secondary fault results when a component is exposed to an operational or environmental condition which exceeds the design rating of that component. A command transition does not involve actual component failure. It simply means that the component is in the wrong state at the time of interest because it was commanded to that faulted state by another faulted component, a human error, or in some cases by an environmental condition.

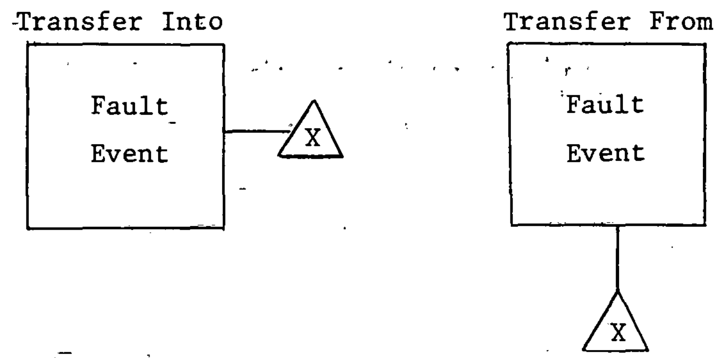
Most of the data available on nuclear components embody both primary and secondary causes for failure; therefore, the distinction between the two types of failure is not made on the fault tree except for the case in which a secondary cause results in multiple component failures, and the distinction is made in code only. The procedure for screening secondary failures for common mode failures is discussed in Section IV-9.

4. GATE TYPES

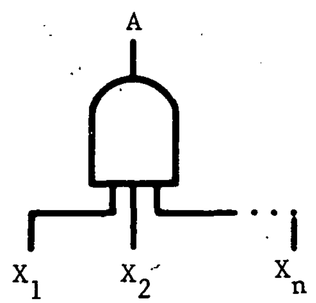
The basic logic gates used in fault tree work are the AND and OR gates. A number of variations of these basic gate types have been introduced in the literature from time to time which are used to handle special situations. Shown in Figure 13 are the standard AND and OR gates and the tabulation OR gate which is introduced here to enumerate a set of fault events which are associated with a series arrangement of components. Safety systems are typically comprised of redundant subsystems each having numerous components connected in series. A fault tree constructed for one of these systems will have, then, a large number of OR gates each with several inputs. The advantage of the tabulation OR gate is that it permits all the fault events within a series arrangement of components to be tabulated rather than being spread out, sometimes over several pages, within individual fault symbols connected together by OR gates.

5. TRANSFERS

Most system fault trees, even the abbreviated form discussed herein, may extend to more than one sheet of paper. To facilitate the extension of a fault tree branch from one sheet to another, transfers are used as follows:



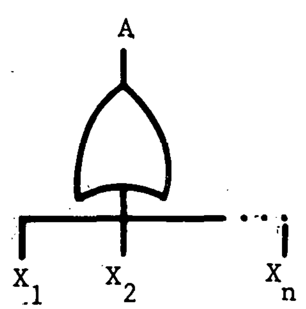
The transfers are arbitrarily lettered or numbered to facilitate cross reference.



AND GATE

The output event A occurs when input events X_1 and X_2 and X_n coexist.

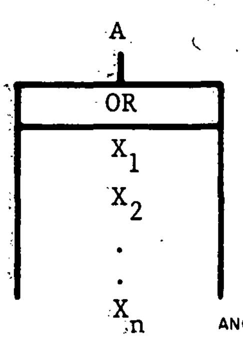
$A = X_1 \cdot X_2 \ldots X_n$ (All input events independent)



OR GATE

The output event A occurs when any one or more input events X_1, X_2 .... X_n exist.

$A = X_1 + X_2 + \ldots.\ X_n$ (All input events independent)



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TABULATION OR GATE

The output event A occurs when any one or more input events X_1, X_2 ... X_n exist.

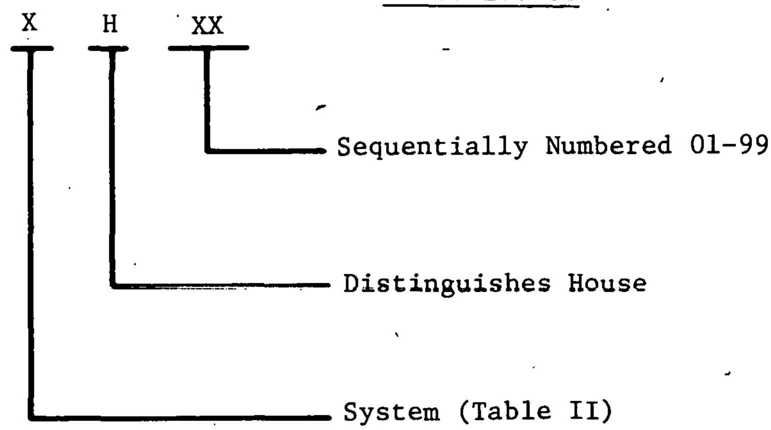
$A = X_1 + X_2 + \ldots\ X_n$ (All input events independent)

Fig. 13 Abbreviated fault tree logic gates.

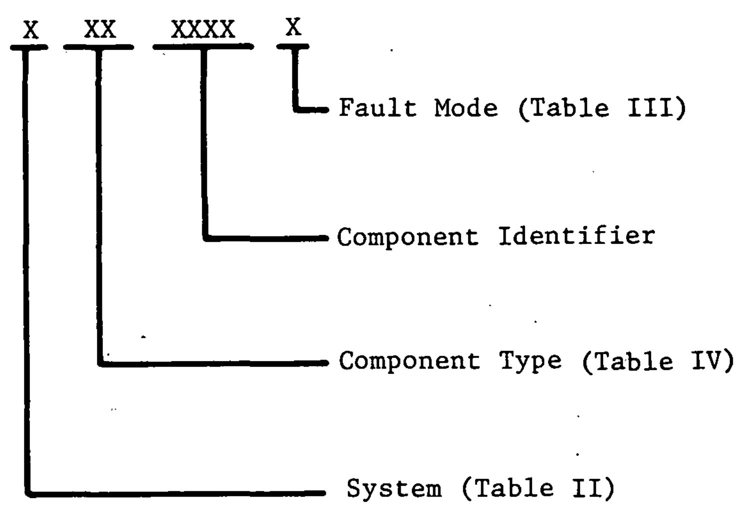
6. EVENT NAMING

In order to facilitate the computer handling of events, each basic input event (events which are not expanded into their respective causes) is given a code name. The identification codes are presented in Tables II through V. Required conditions, or "house" events, are identified on the model by a four-character naming scheme as follows:

House Events



Fault Events are given an eight-character name described as follows:



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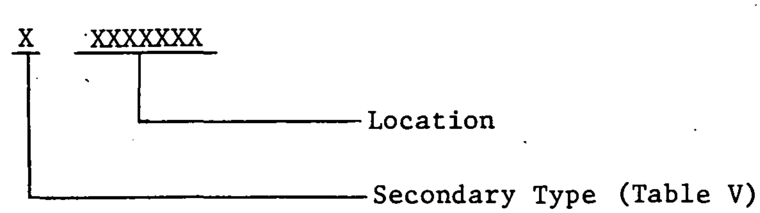
The component identifier in the code is identifiable (where practicable) with the name given the component in the facility identification.

Secondary events which are expected to have significant effect on component failure and are suspect of affecting multiple components (common mode) are given a different eight-character name than that described previously. This secondary event code is characterized by the type of secondary event and location:



The potential secondary event location is best identified by building, room number within the facility, and cabinet number if applicable. If all rooms within the facility are uniquely numbered, the building number is not needed.

All events which are unique in the system must be given a unique name. An event may appear in more than one place on the model or on multiple models, but if it is the same event in the system it must be given the same name.

7. REQUIRED CONDITIONS

A system can assume a variety of possible off, standby, or normal operational states depending on plant conditions and operational requirements. For example, a water pump may be off if the water level in a tank is high but on if the water level is low, a diesel generator may be required to start if the offsite power fails, or a valve may be required to close if a fault has occurred in a downstream component. In fault modeling, inclusion by the analyst of the conditions upon which a system or component is required in the analysis is important. A system fault is not considered a fault unless the system is required. For example, failure of a diesel to start at any time other than when the diesel is needed is not a fault insofar as the analysis is concerned.

Required conditions in a fault tree analysis can be in the form of explicit assumptions and the fault tree constructed accordingly, or the required conditions can be incorporated directly into the fault model. The latter is preferred because it provides versatility in the use of the model. When incorporated

TABLE II

SYSTEM CODE

Code	System
A	Surge tank
B	Core deluge
C	Demineralizer
D	Primary coolant
E	Auxiliary flow
F	Secondary coolant
G	Reactor
I	Reactor protection
J	Degassing
K	Demineralizer
M	Experiment

TABLE III

FAULT MODE CODE

Code	Fault Mode
A	Does not start
B	Open circuit
C	Close valve
D	Does not open
E	Engaged
F	Loss of function
G	Disengaged
K	Does not close
L	Leakage
M	Exceeds limit
N	No input
O	Open valve
P	Plugged
Q	Short to power
R	Rupture
S	Short to power

into the model, required conditions are shown within the "house" symbol. The "house" serves as a switch to turn on those events which are faults when the required conditions exist and off when the required conditions do not exist. The "house" is input into one input of an AND- gate, and the subtree of faults is input into other inputs of the AND gate as shown in Figure 14.

In some situations, to turn on or off subtrees by connecting the "house" to the input of an OR gate is desirable before going into an AND gate as shown in Figure 15. In this case the required condition is inverted (stated negatively) such that when the "house" statement is true the AND gate is enabled; when the "house" statement is false only the existence of faults described by the associated subtree enable the gate. Typically this inverted logic arrangement is used in fault modeling standby redundancy.

The house is also used to describe mutually exclusive faults, in which case two "houses", as shown in Figure 15, are used – one or the other house can be on but not both at the same time.

The house is also frequently used to classify faults for which each fault classification results in a different consequence. For example, in the evaluation of a reactor containment classification of breach areas (faults) according to size may be desirable as shown in Figure 16. In the computer evaluation of the preceding fault tree either or both houses may be turned on depending on whether the analyst is interested in faults <2 in.^{2}, >2 in.^{2}, or all faults, respectively, where the faults in each

TABLE III (contd.)

FAULT MODE CODE

Code	Fault Mode
T	Test and maintenance outage
W	Does not actuate
X	Operational or maintenance fault

TABLE IV

COMPONENT TYPE CODE

Mechanical Components

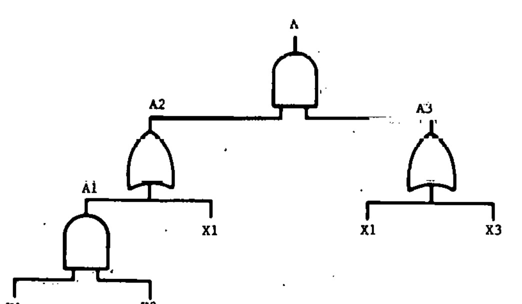
Code	Component
AC	Accumulator
AV	Valve, air operated
BL	Blower
CD	Control rod drive unit
CP	Pipe cap
CV	Check valve
DL	Diesel
FE	Flow element
FL	Filter or strainer
GB	Gas bottle
GK	Seal or gasket
HE	Heat exchanger
HV	Valve, hydraulic operated
KV	Valve, solenoid operated
MV	Valve, motor operated
NZ	Nozzle
OR	Orifice
PM	Pump
PP	Pipe
PV	Pressure vessel
SV	Safety valve

category are listed under the tabulation OR gate (described in Section IV-4).

8. BOOLEAN SIMPLIFICATION

The final process in developing a fault model of a system to which probabilistic values can be assigned involves removing redundancies from the Boolean expression of the model, usually by using the PREP or MOCUS computer codes[6,7]. The analyst can, however, often save considerable time by the application of the same process in developing the fault model in the beginning. However, the analyst should not necessarily try to reduce the model to its simplest Boolean form as it is being constructed, but knowledge of how the model is simplified will sometimes allow the analyst to construct the model more efficiently.

The process of reducing a fault model to its nonredundant Boolean form requires first that the fault model be transformed into an algebraic expression as illustrated by the following example:



$$
\begin{aligned}
A &= A_2 \cdot A_3 \\
&= (A_1 + X_1) \cdot (X_1 + X_3) \\
&= (X_1X_2 + X_1) \cdot (X_1 + X_3) \\
&= X_1X_1X_2 + X_1X_1 + X_1X_2X_3 + X_1X_3 \qquad (1)
\end{aligned}
$$

TABLE IV (contd.)

COMPONENT TYPE CODE

Mechanical Components

Code	Component
TB	Turbine
TG	Tubing
TK	Tank
TZ	Transmitter
VT	Vent
VV	Valve, relief
XV	Valve, manual
OO	Event not involving component

Electrical Components

Code	Component
AM	Amplifier
AN	Annunciator
AS	Buffer
BC	Battery charger
BS	Bus
BY	Battery
RE	Relay
CL	Actuator controller
CA	Cable
CB	Circuit breaker
CC	Capacitor
CM	Comparator
CN	Converter
CT	Current transformer
DC	dc power supply
DE	Diode or rectifier
FU	Fuse
GE	Generator
GS	Ground switch

The preceding algebraic expression contains "AND" and "OR" redundancies which can be removed by using the following idempotent relations:

$$A \cdot A = A \tag{2}$$

$$A + A = A \tag{3}$$

$$A + AB = A \tag{4}$$

By application of these relations to algebraic Expression (1), the model reduces to $A = X_1$. In this example, the analyst would not expand X_2 and X_3 into their respective causes of failure because the models represented by those variables would disappear in the end result.

9. COMMON MODE FAILURES

Single events that fail components in two or more redundant systems or subsystems are common mode failures. They are events which violate any assumptions of independence of redundant systems. Common mode failures can take the form of design or manufacturing defects which emerge as component failures in a common time frame; systematic human errors in the maintenance, testing, or operation of systems; or unexpected environmental or operational transients which result in multiple component failures.

TABLE IV (contd.)

COMPONENT TYPE CODE

Electrical Components Code	Component
HG	Heating element
HT	Heat tracing
IV	Inverter
KS	Lockout switch
LA	Lightning arrester
LS	Limit switch
LT	Light
ME	Meter
MO	Motor
MS	Motor starter
ND	Neutron detector
OL	Overload, thermal
OT	Transformer
PS	Pressure switch
PT	Potentiometer
QS	Torque switch
SW	Hand switch
TC	Temperature controller
TF	Transmitter
TI	Timer
TM	Terminal board
WE	Wire
00	Event not involving component

Before any attempt is made to search for common mode failures, the system fault tree, as constructed in Section IV-2, should be free of any single hardware or human events that result in system (or multisystem) failure because these failures tend to dominate over common mode event contributions. In other words, common mode events do not need to be identified if single component or human events clearly dominate the system probability of failure.

Common mode failures due to operational and environmental variables are usually identified in fault tree analysis by expanding component failure events into secondary causes for failure. That is, component failure events are expanded to show the potential failure mechanisms which exceed the design ratings of the components. For example, the event "Pipe 1A plugged" in Figure 8 might be expanded into possible causes for failure such as "Pipe 1A plugged due to freezing" or due to any number of possible causes depending on the imagination of the analyst. If, in this example, freezing can plug Pipe 1A and components in the redundant subsystem, freezing would be a potential common mode failure event. To expand the fault tree indiscriminately without some real basis for doing so, however, into secondary events can be extremely time consuming and costly.

TABLE V

SECONDARY EVENT TYPE CODE

Code	Event
C	Freezing
D	Dust
E	Earthquake
F	Fire
G	Wind
H	Humidity
K	Corrosion
M	Missile
P	High pressure
R	Radiation
S	Steam
T	High temperature
W	Flood
X	Pipe whip or hammer
Z[a]	Proximity

[a] Z is a code used to indicate that redundant components, because of their close proximity, are subject to a large number of unknown secondary events not readily classified.

The method proposed herein for treating single environmental causes for multiple component failures requires, first, that the analyst determine the location of each component identified by the fault tree analysis. The location is recorded in the column provided in the fault summary of Table VI. Next he examines each component in its operating location to determine: first, whether any of the secondary events listed as column headings can occur in the component location; and, second, whether, if a secondary event can occur in that location, will the secondary event cause failure of the component. An estimate of the secondary event occurrence likelihood is shown in the upper half of the space provided in the fault summary and an estimate of the likelihood of the secondary event producing component failure is shown in the lower half of the space provided. These need only be order of magnitude probability estimates and are usually written -1, -2, -3, . . ., corresponding to probability values of 10^{-1}, 10^{-2}, 10^{-3}, . . ., respectively. The list of secondary events shown in the fault summary column headings is certainly not complete and should be expanded where appropriate.

Finally, if there are secondary events that have a relatively high likelihood of occurring and causing component failure, they should be named and treated as additional component failure events on the fault tree and on the fault summary. The product of these

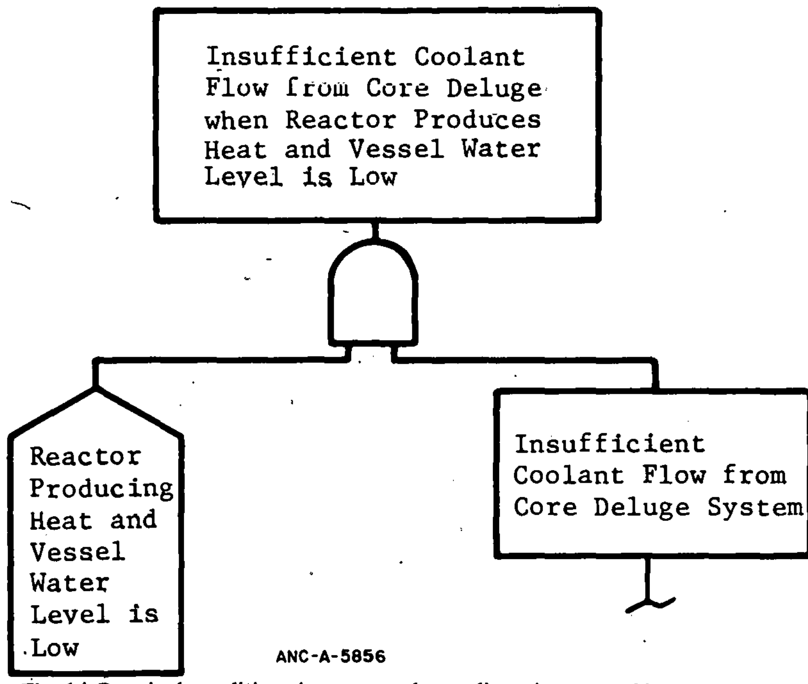


Fig. 14 Required conditions incorporated as a direct input to AND gate.

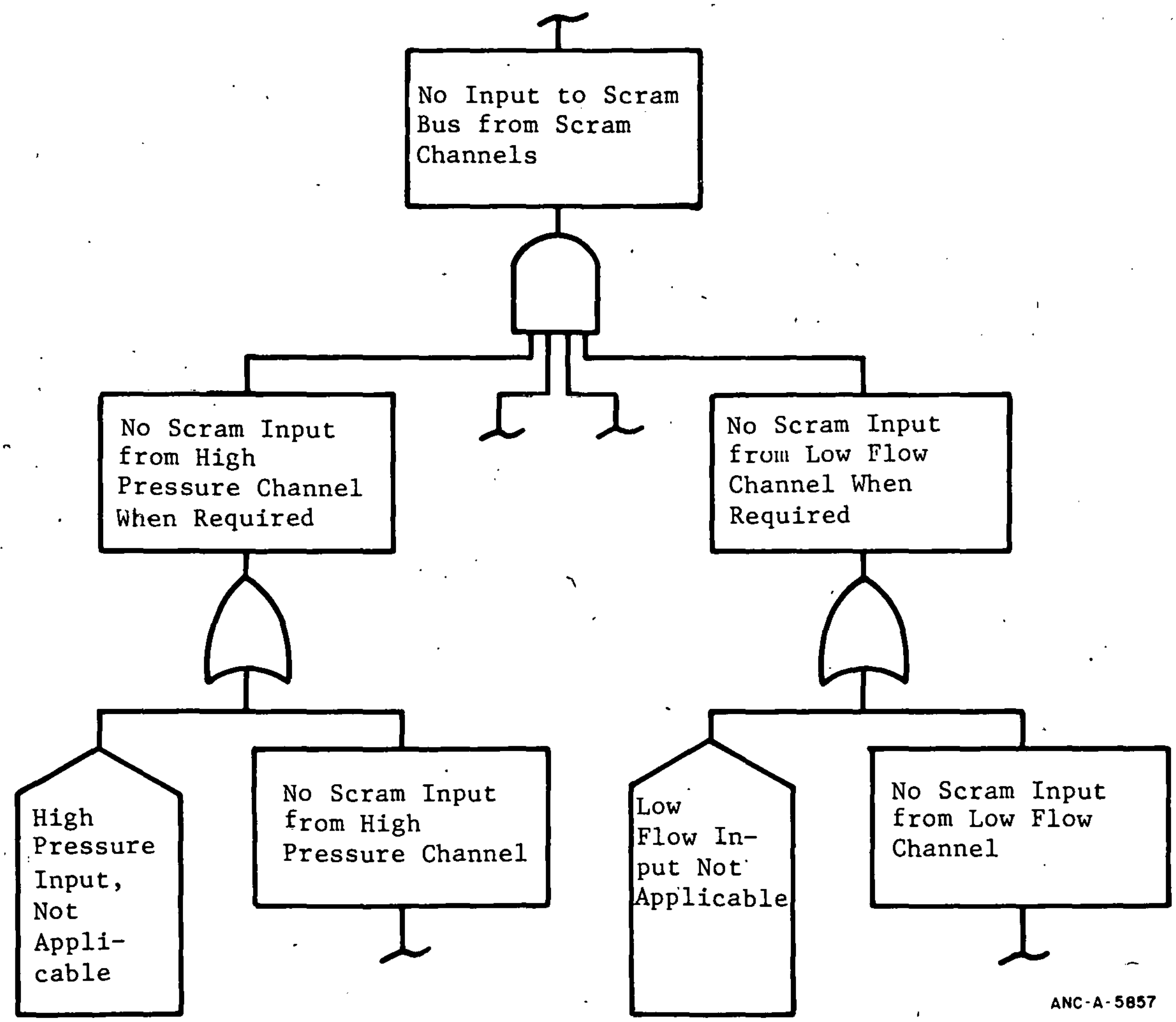


Fig. 15 Required conditions incorporated as inverted inputs to AND gate.

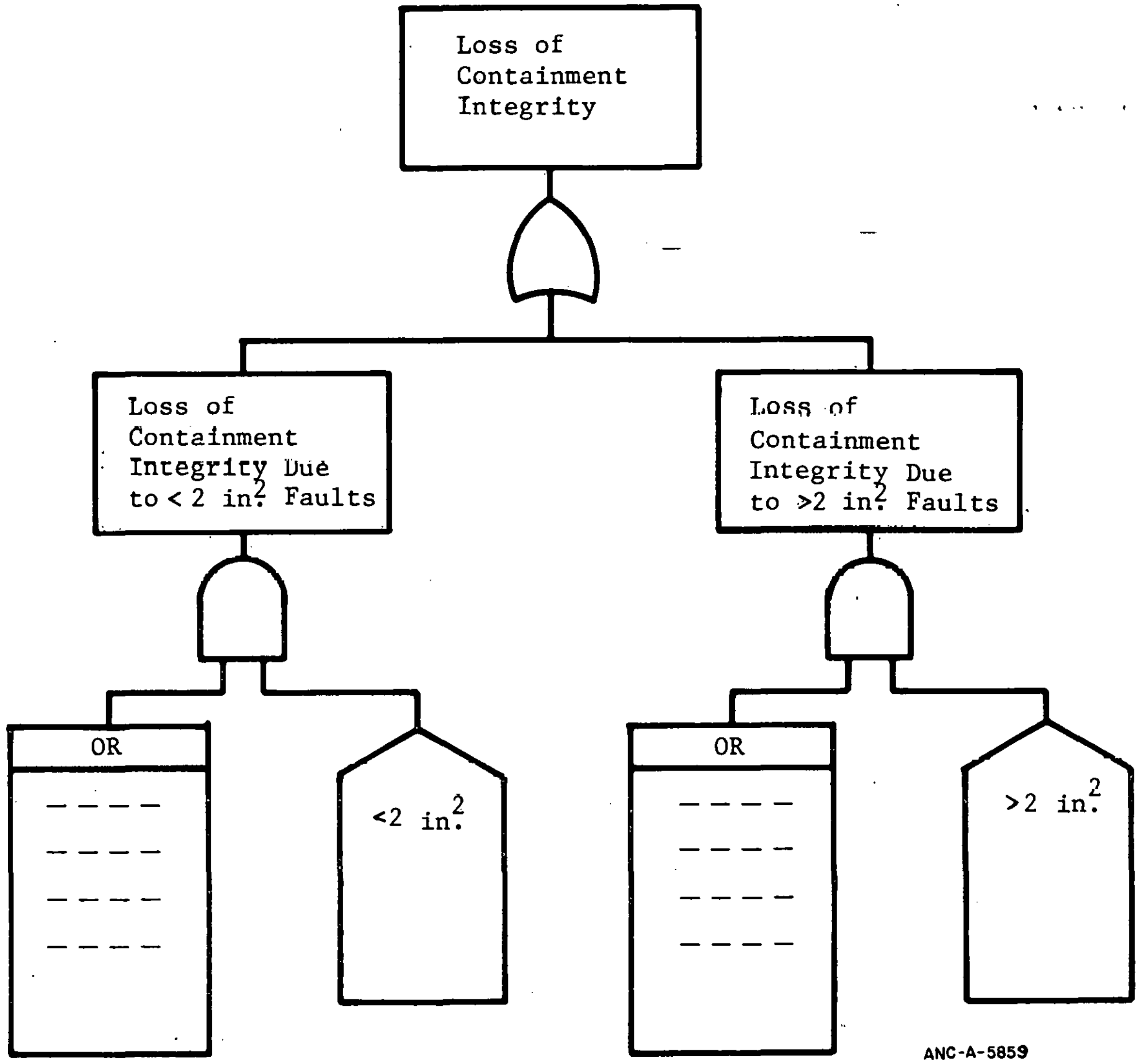


Fig. 16 Classifying faults using the house.

two probabilities, likelihood of occurrence and likelihood of causing failure, should be large compared with other secondary events treated in the same manner. Typically, these events do not take on much significance unless the product is of the order of 10^{-6} and greater. The procedure for treating common modes can be illustrated by examining Table VI[a]. Pipe 1A might rupture due to an earthquake (probability of 10^{-3}), high pressure (10^{-2}), freezing (10^{-1}), missiles (10^{-2}), or pipe whip (10^{-1}). The likelihoods of these events occurring in the pipe tunnel are 10^{-8}, 10^{-8}, 10^{-9}, 10^{-8}, and 10^{-7}, respectively. The products of these probabilities are relatively low (10^{-11}, 10^{-10}, 10^{-10}, 10^{-10}, and 10^{-8}); therefore, they are relatively insignificant contributors. Relay 26A is subject to failure by fire (10^{-2}), dust (10^{-2}), or corrosion (10^{-3}). The likelihoods of fire, dust, and corrosion in Room 211 are 10^{-6}, 10^{-3}, and 10^{-5}, respectively. Dust, which has a combined likelihood of occurrence and causing failure of 10^{-5}, is potentially an important contribution to relay failure; therefore, a new event, "dust" with a code name D0000211, is entered in the fault summary. The code

[a] The values shown are for illustration purposes only and are not intended to be characteristic of ETR.

TABLE VI

COMMON MODE EVENTS ON FAULT SUMMARY

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^{-x}) *																		LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI TEMP	PRESSURE	HUMIDITIY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT			
BPPOO1AR	Pipe 1A	Rupture	10^{-9}	270	10			-8 -3			-8 -2		-9 -1	-8 -2	-7 -1									Pipe Tunnel	
BREO26A	Relay 26A	Doesn't Close	10^{-5}	270	10	-6 -2											-3 -2		-5 -3					Rm 211	
DOOOC2111	Dust			(New Event)																				Rm 211	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

name is also listed on the fault tree under the tabulation OR gate where the failure modes of Relay 26A are shown. D0000211 is a unique identifier for "dust in Room 211". If in the process of applying this procedure to other components, and the event name D0000211 appears in other trees or subtrees representing redundant systems or subsystems, respectively, the event is a common mode event. That is, the event D0000211 would appropriately affect the nonredundant form of the Boolean expression resulting from one or more trees containing the event.

10. HUMAN ERRORS

Human errors are relatively high probability events; therefore, human intervention or human inputs to components are important contributions to the probability of system failure. Switches, valves, adjustment pots, and test plugs are only a few of the many components which are subject to normal human input. All potential human errors should be identified on the fault tree at the component where the human intervention takes place. For example, if the only place a valve can be operated is from a switch in the control room, the human error event would be associated with the switch in the control room and not the valve. If the valve can be operated remotely and locally, then the human error fault events should appear both places.

Human errors are generally classified as errors of commission and errors of omission[1]. Errors of commission are those for which an operator or maintenance man will act inadvertently with a component of the system (for example, an operator throws the wrong switch or a maintenance man misadjusts a limit switch). Errors of omission are those for which the operator forgets to perform a required act (for example, fails to start pump). The type of human error should be clear in the fault statement. For example, the fault statement in the fault summary might read, "operator forgets to start Pump 1B" for an act of ommission, or "valve inadvertently closed" for an act of commission.

11. TEST AND MAINTENANCE

System outages due to tests and maintenance and the human errors which can accompany test and maintenance activities can be important contributors to the risks of nuclear plants. Most systems associated with test reactors are tested and maintenance is performed when the reactor is shut down; therefore, test and maintenance outage, as such, is not an important risk factor. However, where on-line testing and maintenance has been provided in the design, a system which is redundant can change to a nonredundant system during the time tests and maintenance are performed unless override features have also been provided in the design.

Outage due to test or maintenance is treated on the abbreviated fault model by showing an additional component fault event on the fault tree and on the fault summary for any subsystem or portion thereof which is unavailable during test and maintenance. Although not a fault in the strict sense of the word, outage is treated as a component fault with a mode designation "test" or "maintenance" and a fault mode code designation "T". Unless each component is tested or maintained separately and at different times, only the component requiring the longest outage time is shown as a fault time. If each component is tested or maintained separately and at different times, each component should be treated as a test and maintenance fault.

If a valve or other component can be left in the wrong state as a result of a test or maintenance error, the fault is also shown on the fault tree and is treated as a human error as discussed in Section IV-10.

12. ANALYSIS STAGING

The abbreviated fault tree analysis described in the preceding section helps the analyst to stage the analysis effort. That is, he can determine the overall logic structure of complex systems and multiple systems first before performing a detailed examination of components within the system. Thus, staging allows the analyst to identify the more important, or critical, paths of the system without wasting time on details which may, in the end result, be unimportant. To stage the analysis, the analyst constructs the abbreviated fault tree without identifying the individual events normally listed under the tabulation OR gate. Instead, each tabulation OR is treated as a single component until the fault tree is reduced to its nonredundant Boolean form. Then, only those tabulation OR gates which appear as critical cut sets in the nonredundant Boolean form are expanded to include individual component events.

Caution should be exercised regarding the analysis staging just discussed: First, the tabulation OR gates must be independent of other tabulation OR gates (they should not contain common elements if expanded), and, second, reliance on the importance of tabulation OR gates resulting from staging can ignore potential significant common mode events among those individual component fault modes not included, particularly if the staging effort does not produce single component events that can result in system or multiple system failure. (In most cases single hardware and single human error events will dominate over the contribution by any common mode events.)

13. DATA

Once all basic fault events have been identified for a system, probability values are assigned to, or information is provided in order to calculate probability values for, individual event entries listed in the fault summary. Component failure probabilities are calculated from failure rates and appropriate time intervals (or cycles if the failure rate is cyclic). Component failure rates which are time dependent are usually expressed in failures per 10^6 hours; component failure rates which are related to the number of times the component is operated are expressed in failures per 10^6 cycles. For most standby systems a cyclic component is usually called upon to operate once; therefore, the failure probability on demand is also the failure rate (1" is entered in the fault summary time column).

The time intervals used to calculate failure probabilities for components in nonoperating standby systems, typical of most safety systems, are the fault duration times. Usually this duration is the average time during which a fault can exist without being detected plus the time to repair the component when the system is required to stand by. If a component failure can only be detected by periodic tests, then the average fault duration time is one-half the test interval plus any time needed to repair the component. One hour is usually used as an arbitrary minimum in the calculation if the component status is continuously monitored. The repair time is the lesser of the times required to actually repair the component or to return the reactor to a state in which the system is no longer required.

Probability values for individual events shown on the fault summary pertaining to test or maintenance outage are determined by taking the accumulated total test or maintenance outage time and dividing by the time the system is required.

14. QUANTITATIVE EVALUATION OF FAULT MODELS

The evaluation of a fault model basically involves two steps: (a) qualitative analysis and (b) quantitative analysis.

Qualitative evaluation of fault models is the conversion of the failure logic to its nonredundant Boolean form. For any individual TOP event of concern, all sets of necessary and sufficient individual events that cause this TOP failure must be identified. Any set of necessary and sufficient events is called a minimal cut set. All failure modes of a TOP event of a system are completely described by the collection of all minimal cut sets. The computer codes now available, and which are receiving the greatest use for this type of qualitative evaluation, are PREP[6] and MOCUS[7].

The PREP computer code for determining minimal cut sets allows one of two different options, deterministic testing and Monte Carlo simulation. The deterministic testing in theory essentially checks all possible combinations of individual events to determine whether their failure will result in the TOP event. The Monte Carlo approach weights the individual events, thus accelerating the process of determining the minimal cut sets. However, with the Monte Carlo approach, the analyst can not be sure that all the minimal cut sets are identified.

The algorithm used in MOCUS to find the minimal cut sets starts with the TOP event failure and proceeds to the individual failures in order to determine the minimal cut sets. Basically, the MOCUS algorithm builds a matrix along the following lines: the fault model gates are randomly labeled with a value a_i and the individual events are randomly labeled with a value b_i; the first element in the matrix is set to the value of a_i representing the gate immediately under the TOP failure. From this point on, the object is to eliminate all the gate a_i values from the matrix such that only the b_i values for the individual events remain. Each row of the matrix is now a cut set. The rows are then searched to determine the minimal cut sets.

The inputs to both the PREP and MOCUS computer codes are compatible. For large component minimal cut sets, the MOCUS code requires much less computer execution time. Usually the smaller component cut sets consisting of possibly only one, two, or three individual events describe the majority of the failure modes quantitatively; therefore, either PREP or MOCUS may be used.

The qualitative analysis of the fault model thus provides the framework for the probabilistic or quantitative evaluation. With the modes of failure obtained from the qualitiative analysis in the form of minimal cut sets and the reliability characteristics of individual events, the TOP event or system reliability characteristics can be calculated. The time-dependent failure rate or hazard rate and the fault duration time for each individual event completely describes the reliability characteristics necessary for the quantitative evaluation of the system. From these minimal cut sets, hazard rates, and fault duration times, the reliability characteristics associated with system failure or operation can be calculated. The computer codes now in greatest use to calculate these are KITT[8] and SAMPLE[1].

The KITT computer code is an application of Kinetic Tree Theory[8] and calculates the complete spectrum of reliability characteristics. The SAMPLE computer code evaluates the unavailability for the fault model specified by the minimal cut sets and also calculates confidence limits for the unavailability.

The input-output for the PREP, MOCUS, and KITT computer codes are totally compatible. The previously mentioned computer codes are written in FORTRAN IV for the IBM 360/75 computer. The assumptions and limitations for each code are given in their documentation.

V. DEFINITIONS

1. Accident Initiating Event – A parameter which enters a region where protective action is required in order to prevent an accident.

2. Fault – An event which can, either by itself or in combination with other events, result in a system malfunction.

3. Failure – A component or system fault which is nonreversible; that is, repair is required.

4. Common Mode Event – An event originating within or external to a system that causes multiple component or system malfunctions.

5. Direct Input – An event which, if it occurs, has a probability of unity of producing a failure.

6. Coupled Input – An event which, if it occurs, has a probability less than unity of producing component failure.

7. Secondary Event – An event which exceeds the design rating of a component.

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APPENDIX A

RISK ASSESSMENT OF ETR LOSS OF FLOW EVENT

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APPENDIX A

RISK ASSESSMENT OF ETR LOSS OF FLOW EVENT

This appendix illustrates the risk assessment methodology by applying it to the Engineering Test Reactor (ETR) located at the Idaho National Engineering Laboratory. This assessment is done by progressing through the various steps of a risk assessment using a loss of forced flow as an example accident initiating event.

1. DISCUSSION OF ETR

The ETR was completed in 1957 to fulfill a demand for more and larger facilities to provide higher neutron fluxes and more spaces for larger samples than existed in previous test reactors. Its purpose is to expose samples of reactor fuels, coolants, and structural materials to intense neutron fluxes produced in or near the reactor core. The samples are then studied to determine possible changes in their material properties. Items that are irradiated in the reactor typically include comparatively large reactor components proposed for use in other military, commercial, or research reactors. When tested, these components are inserted directly in spaces provided in the core region or in dynamic loops, that is, closed systems in which a coolant circulates around the experiment at flows, temperatures, and pressures that are independent of coolant in the reactor.

The design power level of the ETR is 175 thermal megawatts. The reactor is capable of producing a maximum thermal neutron flux of $8 \times 10^{14} \frac{n}{cm^2\text{-sec}}$ and a maximum above-thermal neutron flux of $2.5 \times 10^{15} \frac{n}{cm^2\text{-sec}}$. Figure A-1 is a simplified diagram of the ETR.

Recently the ETR has undergone major modifications to accept the Sodium Loop Safety Facility (SLSF). One of the purposes of this experiment is to study the characteristics of fuel element failure propagation in a fuel bundle cooled by liquid sodium as the primary coolant and helium gas as the secondary coolant.

In preparation for ETR operation with the sodium loop, the instrumentation associated with reactor control and the plant protection system has been improved. In general, the improvement consists of the replacement of much of the older vacuum tube circuitry and mechanical relays with integrated circuitry and other more reliable devices. The SLSF installation has its own plant protection system whose function is to shut the reactor down in the event experimental variables exceed predefined limits.

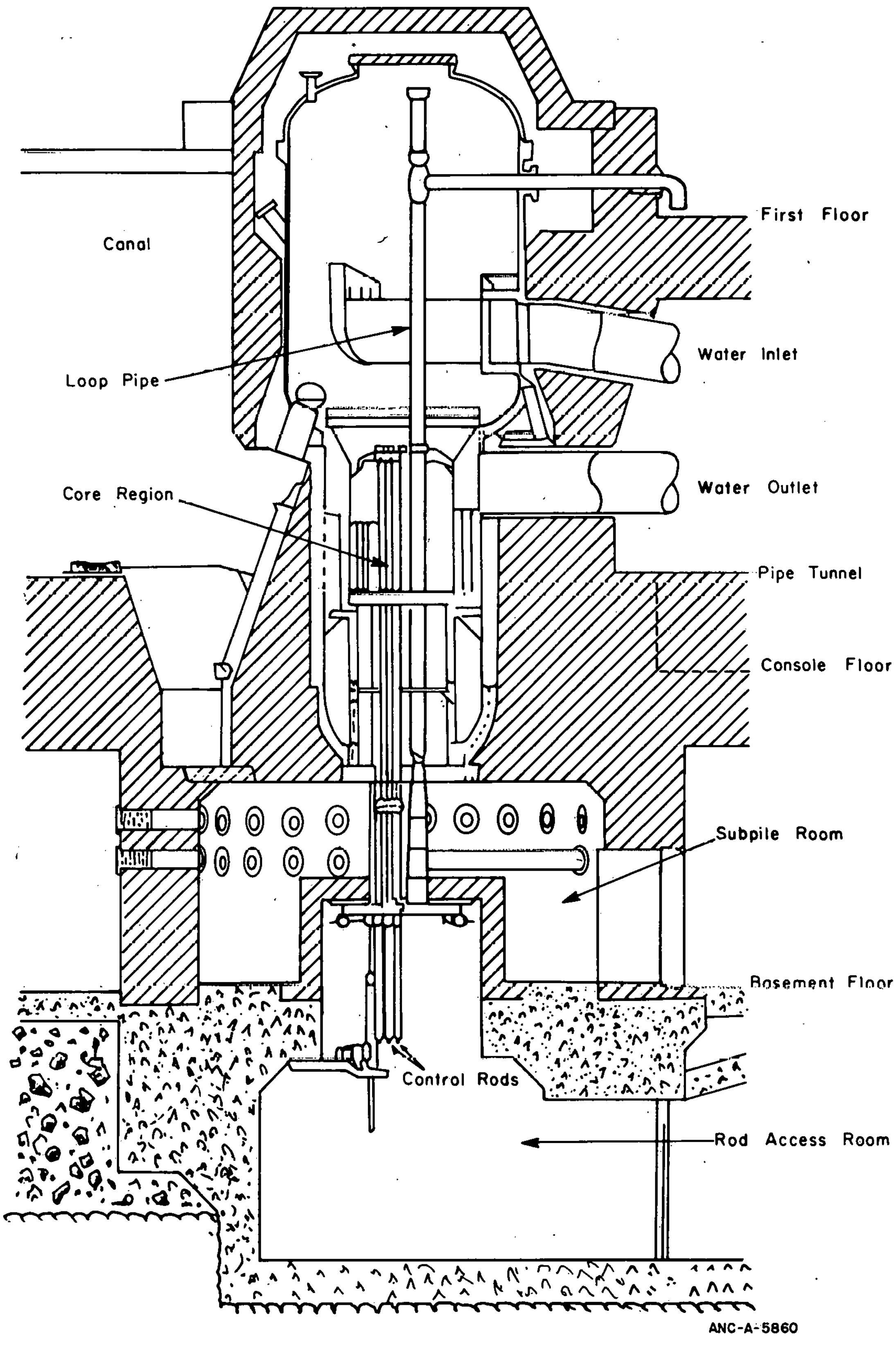

Fig. A-1 Simplified diagram of ETR.

2. OVERVIEW OF DISCUSSION

The first task considered in the sample risk assessment was to identify all accident initiating events that might occur which could, in the absence of mitigating action, lead to uncontrolled release of fission products. This accident initiating event identification analysis is illustrated in Section 3. The loss of flow accident is discussed in Section 4. Event tree sequence probabilities were obtained by the evaluation of individual systems which appear in the sequences. These individual systems evaluations are covered in Section 5 and are combined for sequences in Section 6. Sections 7 and 8 discuss the analysis needed to determine the core damage and consequences resulting from each loss of flow sequence. Finally, the results of the risk assessment are discussed in Section 9.

3. IDENTIFICATION OF ACCIDENT INITIATING EVENTS

The procedure described in the report has been used to identify the accident initiating events for the ETR. The study is not complete but has been developed as far as necessary to illustrate the procedure described in the report. The example considers only the events that would lead to radioactivity releases to the atmosphere. Other energy releases (for example, steam) were not considered. The top level fault tree showing the logic for the accident initiating events is shown in Figure A-2.

3.1 Undesired Radiological Release

A broad statement for the top event of the fault tree was selected so that the undesired radiological release would include all levels of release. The top event, then, contains a statement about all possible releases of radioactivity from the ETR.

3.2 Release Sources

All sources of radioactive material at the ETR are shown at the next level of the fault tree. A release from any one of these sources would result in the undesired radiological release stated in the top event of the fault tree. Because the ETR is one of several reactors in the Test Reactor Facility, some storage locations are shared with other reactors. The fault tree includes all sources associated with the ETR even though shared with other reactors.

The identification of all sources was helped by a review of a nuclear material accountability list in an unpublished monthly report. This report identifies the type, amount, and location of radioactive material for most of the sources listed on the fault tree. Other sources were found by a review of the operating system of ETR.

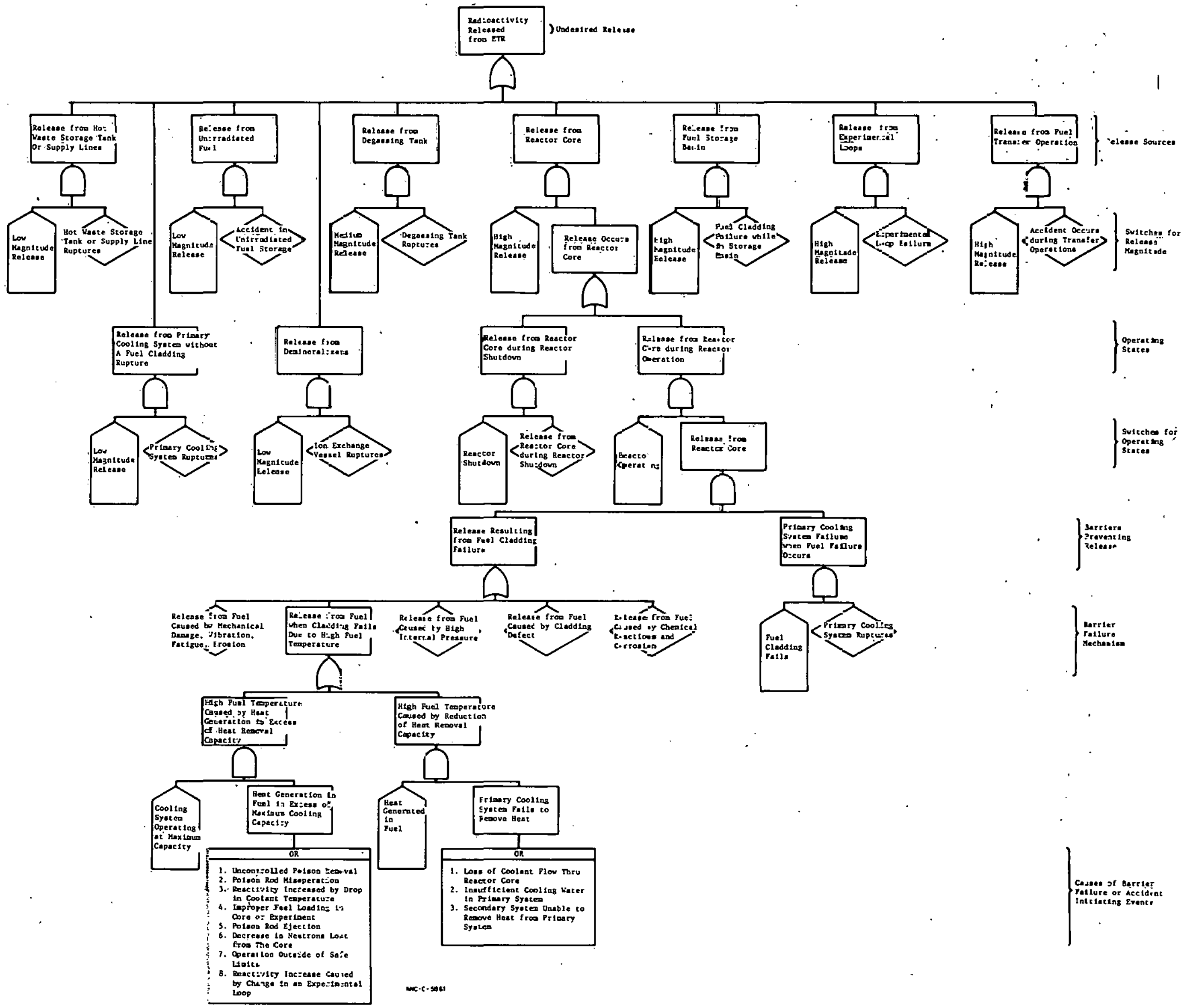

Fig. A-2 ETR accident initiating events fault tree.

3.3 Switches for Release Magnitudes

Because the top event does not specify a specific release magnitude, all levels of release are included on the tree. The potential magnitude of release from each source has been assigned as high, medium, or low. If a study of a specific release magnitude were to be desired, the tree below the desired magnitude statements would be included. For this example study, the tree for a release from the reactor core is the only one developed. Each fault in the diamond symbols would be further developed for a complete study. The switch function of the house symbol was explained in the methodology section in the body of this report (Section IV-7).

3.4 Switches for Operating States

When the operating state of the reactor will affect the radiological release, switches are provided so that the state under consideration can be analyzed. For this example, only the state when the reactor is operating has been developed.

3.5 Barriers Preventing Release

All sources of radioactive material are prevented from release by one or more physical barriers. The barriers that have to be penetrated before radioactivity is released from the fuel in the core are the fuel cladding and the primary cooling system. Both barriers have to be penetrated. The radioactive material in the core is not released if contained by either one.

3.6 Barrier Failure Mechanisms

The next level of the fault tree lists all mechanisms that can cause failure of the barriers. The failure of the primary cooling system has not been developed in this example but would include all ruptures of primary system pipe, valves, pumps, . . ., or the reactor vessel.

All conceivable mechanisms that can cause failure of the fuel cladding are listed at this level. Of these, only the failure caused by high fuel temperature is developed for this example. High fuel temperature can be brought about by two general categories of failure: (a) heat generation in the fuel that exceeds the maximum cooling capacity of the cooling system, and (b) reduced primary cooling system capacity.

3.7 Accident Initiating Events

The events that can bring about the two general categories of failure previously identified are considered to be accident initiating events. Should any of these events occur, reactor protection systems are required to respond to prevent the release of radioactive material from one or both of the physical barriers. Further study of the events will lead to an investigation of all hardware in the reactor systems that can, alone or in combination with other events, fail and cause a release. This investigation will be done by a systems analysis as described in Section 5 of the report. For completeness, the list has been reviewed by individuals knowledgeable of the ETR systems and their operation. The loss of coolant flow accident will be developed as an example.

4. LOSS OF FLOW ACCIDENT

An event which causes loss of coolant flow through the reactor is considered important because it jeopardizes the integrity of the fuel cladding. That is, fuel cladding temperatures can occur which result in melting of the cladding and, ultimately, to release of radioactivity to the atmosphere. The reactor, when operating at power, is a source of considerable heat which must be constantly dissipated by the flow of large volumes of water through the flow passages in the core. Under normal operation, this flow is controlled at about 43,000 gallons per minute, with a reactor vessel inlet temperature of about 120°F. The outlet water temperature is maintained at about 135°F. Upon loss of flow and no scram, the system temperatures will increase to a point at which coolant voiding occurs at the cladding surfaces. The cladding temperatures then commence to rise rapidly to a point (about 1,200°F) at which fuel cladding melting occurs.

Upon loss of flow, the plant protective system and the emergency flow system respond to shut the reactor down and to provide for minimum circulation of coolant, respectively. The plant protective system responds upon (a) low or high differential pressure across the reactor, (b) reactor low inlet-to-outlet pressure, or (c) low flow to cause rapid insertion of all poison control rods into the core. The emergency flow system responds (one of the two pumps is always running when the reactor is operating) to cause a minimum of coolant circulation through the core.

The event tree for the loss of forced flow accident in the ETR is shown in Figure A-3. The event tree is explained as follows:

(1) The ETR event tree starts with the assumption of complete loss of forced primary coolant flow through the reactor with a probability of occurrence of P_1. The first event node, second column, has two branches that represent whether or not the automatic reactor shutdown system is available to scram the reactor. A probability of P_2 that the system is not available (lower branch) to shut down the reactor and a probability of $(1-P_2)$ that it is available to shut down the reactor (upper branch) are assigned.

If the PPS fails to respond, and the reactor continues to operate at power with no flow of coolant, melting will occur. (The reactor will produce fission heat until sufficient coolant voiding occurs to terminate the fission process.) The probability of this series of events, S_3, is $P_1 \times P_2$ (provided P_1 and P_2 are independent).

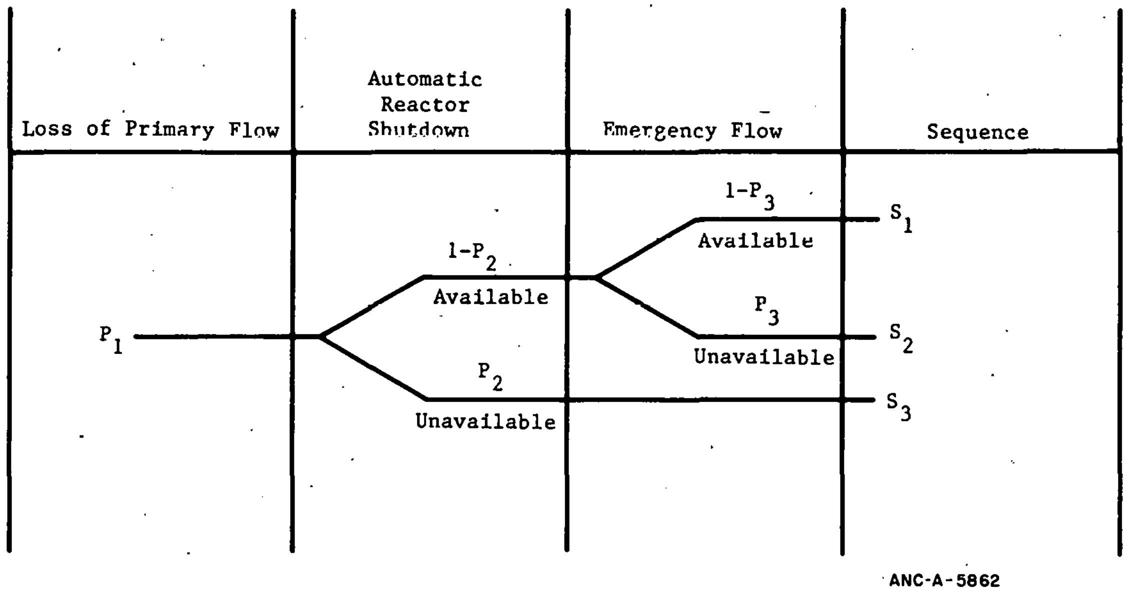

Fig. A-3 Event tree for loss of flow event.

(2) The second event node, third column, is the availability of the emergency flow pumps for operation. Either of the two pumps will supply sufficient coolant for a short period of time to prevent voiding after a loss of flow, where one pump is capable of providing 2,000 gpm coolant flow. The probability of emergency flow unavailability is designated P_3; the availability is $(1\text{-}P_3)$. The sequence probability cannot be determined by a direct calculation of the individual probabilities because the probability values are not independent.

(3) Each sequence shown on the event tree produces a consequence which when multiplied by the preceding sequence probabilities produces risk values for the loss of flow event.

5. SYSTEMS ANALYSIS

The analysis of the loss of coolant flow accident, selected as the accident initiating event for this study, requires an evaluation of the primary coolant system, to determine how it functions, and to identify those malfunctions of the system that could initiate a loss-of-flow accident.

5.1 Primary Coolant System

The primary coolant system removes heat from the reactor core and rejects it to a secondary coolant system which, in turn, dumps it to the atmosphere through cooling towers. A number of accident initiating events can stem from the primary coolant system, for example, loss of coolant, loss of coolant flow, loss of primary system pressure, insufficient heat rejection to the secondary system, . . ., none of which is independent. In this example risk evaluation, determining the probability and potential consequences of the loss of flow event is the only concern, and particularly in this section to determine the probability that a loss of flow will occur at the ETR of sufficient duration to demand a response by the plant protection and auxiliary flow systems.

5.1.1 System Description. The primary coolant system is comprised of the reactor vessel, pumps, valves, heat exchangers, and piping arranged as shown in the simplified diagram, Figure A-4. Coolant flow through the reactor core is maintained at the desired level by series Control Valve P-8 for coarse control, and bypass Control Valve PCV-4-4 for fine control. Flow is controlled on differential pressure across the reactor and outlet coolant temperature. Control Valve P-8 is equipped with mechanical stops which prevent closure beyond 45 degrees. Control Valve PCV-4-4, if completely open, will not permit sufficient bypass flow to prevent removal of decay heat with an emergency flow pump (2,000 gpm) operating. Three of four primary coolant pumps will be used for the SLSF tests.

The temperature of the coolant (about 135°F exit temperature at $\leqslant$177 psig during operation) is sufficiently low at all pressures above atmospheric that a break in one of the primary coolant pipes will not uncover the core. Also, the elevation of the piping which enters the vessel is sufficiently high to prevent draining, and sufficient bypass exists internal to the vessel to prevent any siphoning of water from the vessel in the event of a primary coolant pipe break.

Primary system pressure is maintained at 177 psig by a degassing system which constantly removes coolant from the primary coolant system and, after removing any gas, returns it to the primary system at a constant 177 psig. A surge tank is provided to absorb any transients which may occur in the system. A fraction of the primary system flow is circulated through a demineralizer for removing impurities and maintaining a desired pH. Relief valves are located on the input manifold of each of the four heat exchanger banks.

5.1.2 System Analysis. The primary coolant system was evaluated to identify all events related to the system and dependent systems that could result in loss of forced flow. Any loss of flow was considered whether affecting a part of the core or the entire core. That portion of the electrical power distribution system which supplies power to the primary coolant system was also included in the analysis.

(1) Assumptions. For the purposes of the analysis the following assumptions were made and ground rules were established:

(a) The loss of flow accident occurs when the reactor is operating, and any three of four primary coolant pumps are sufficient and necessary for core cooling.

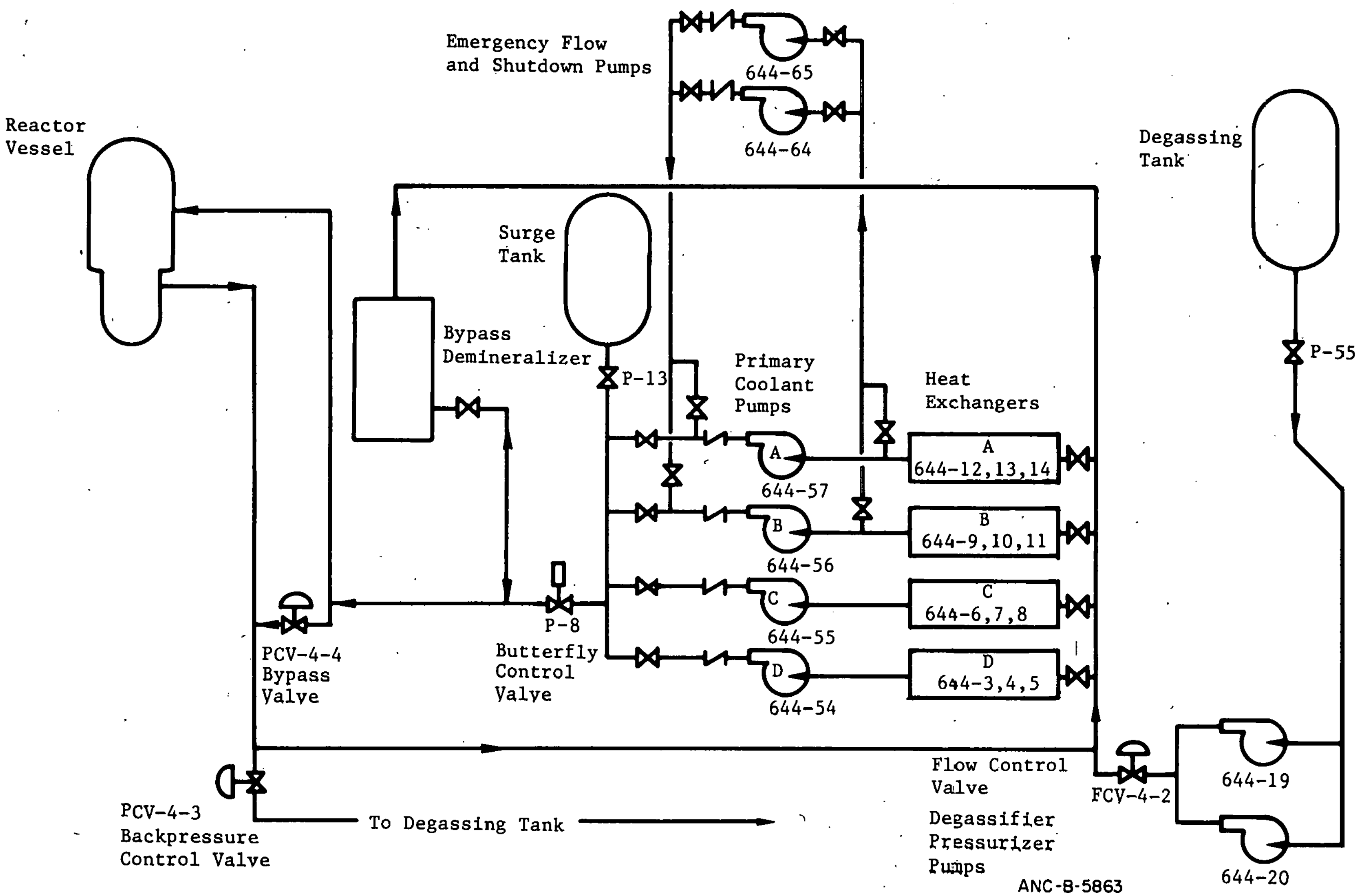

Fig. A-4. ETR primary coolant system - simplified flow diagram.

(b) The nature of the faults produces loss of flow but does not necessarily result in other initiating events; that is, it does not depressurize the system or does not uncover the core (The fault tree developed for the system, however, identifies numerous events that can result in more than just loss of flow.).

(c) Once loss of flow occurs, flow is not restored; that is, the sequences on the event tree are carried to completion. For example, a valve failure could not be repaired in time to prevent the course of an accident initiating event as described by the event tree.

(d) The only responses to the accident initiating event are those described by the event tree.

(e) An operating time for the reactor is 720 hours.

(2) Fault Tree and Fault Summary. The fault tree and fault summary developed for the loss of flow event at the ETR are shown in Figure A-5 and Table A-I, respectively. Since the primary system is a normally operating system (as opposed to an engineered safety system), numerous single events can occur that would result in loss of flow. The fault tree identifies a number of component ruptures that could result in loss of flow. These ruptures, should they occur, would require a response by the core deluge system rather than the emergency flow system as shown on the event tree. That is, a separate event tree would be required to reflect the possible sequences resulting from a loss of flow caused by component ruptures. The rupture events, if included, however, would not alter the quantitative results because the rupture probabilities are small compared with loss of pumping power and human error events identified in the analysis.

Data shown in the fault summary, as is the case with other systems analyzed in this sample risk assessment, were extracted from the Reactor Safety Study[A-1].

(3) Drawings. Engineering drawings were used in the analysis of the primary coolant system. The simplified sketch shown in Figure A-4 was taken from the drawings.

(4) Data. Failure data used in the evaluation were those contained in WASH-1400[A-1].

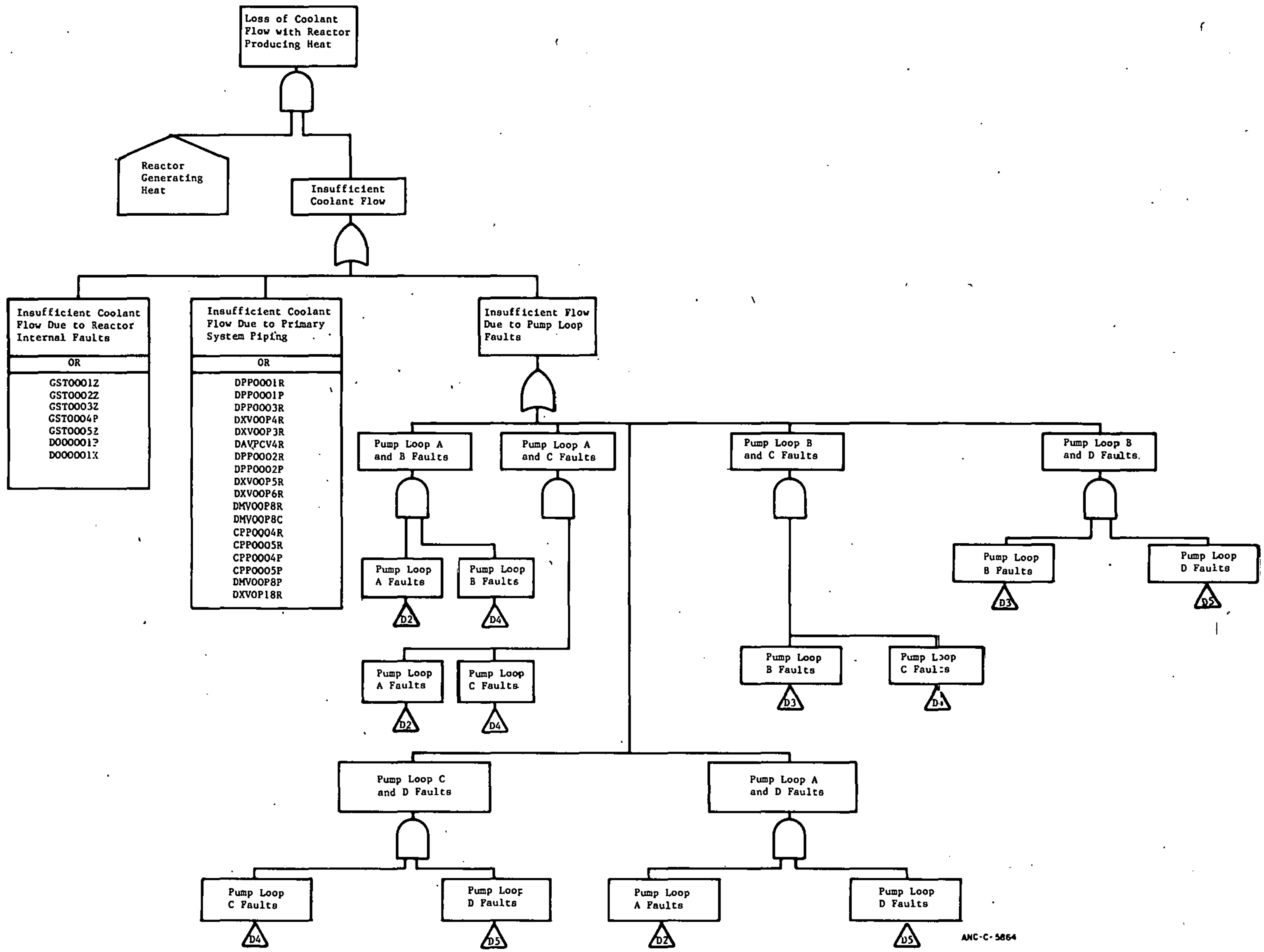

Fig. A-5 ETR loss of forced flow.

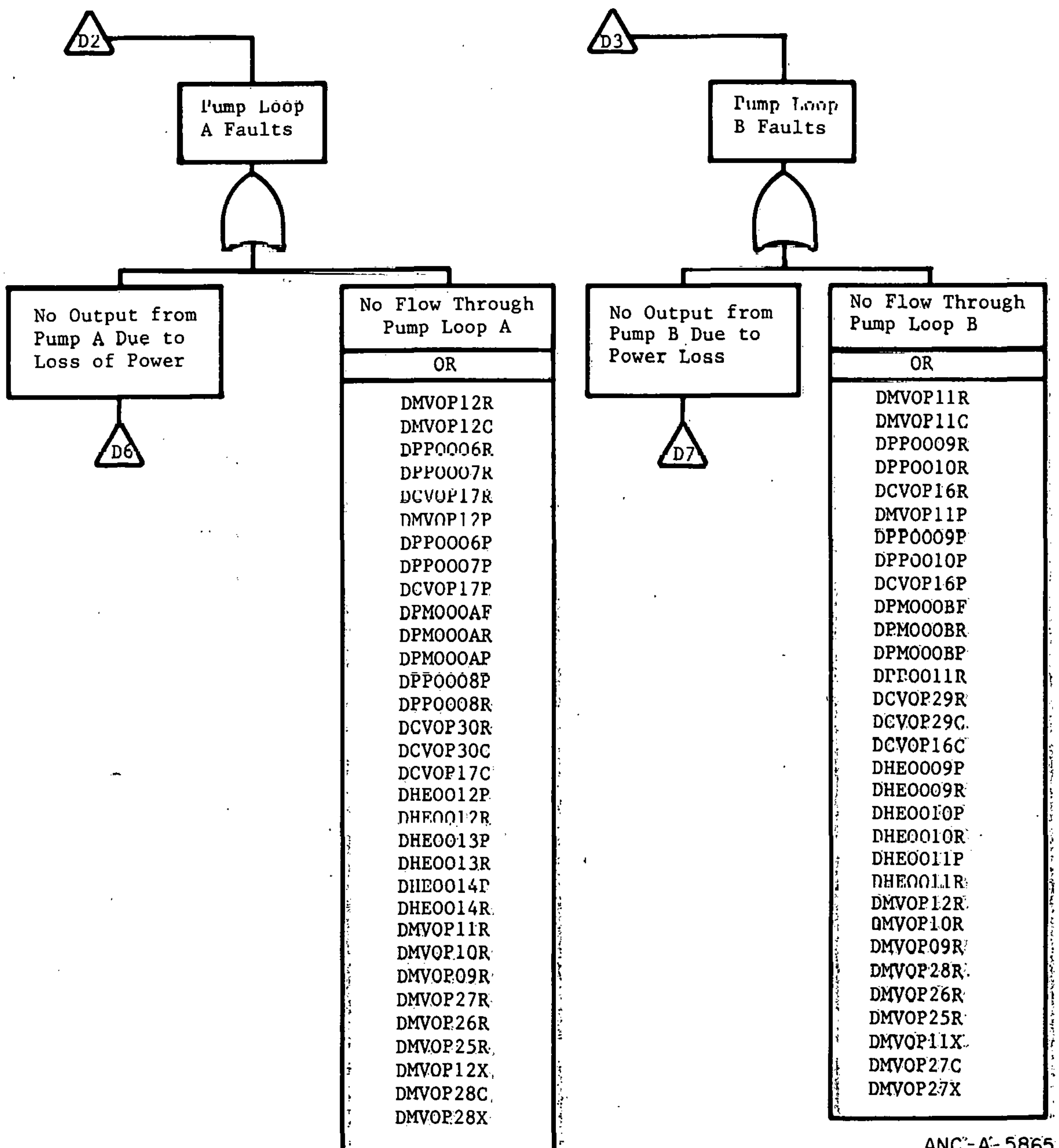

Fig. A-5 ETR loss of forced flow (contd.)

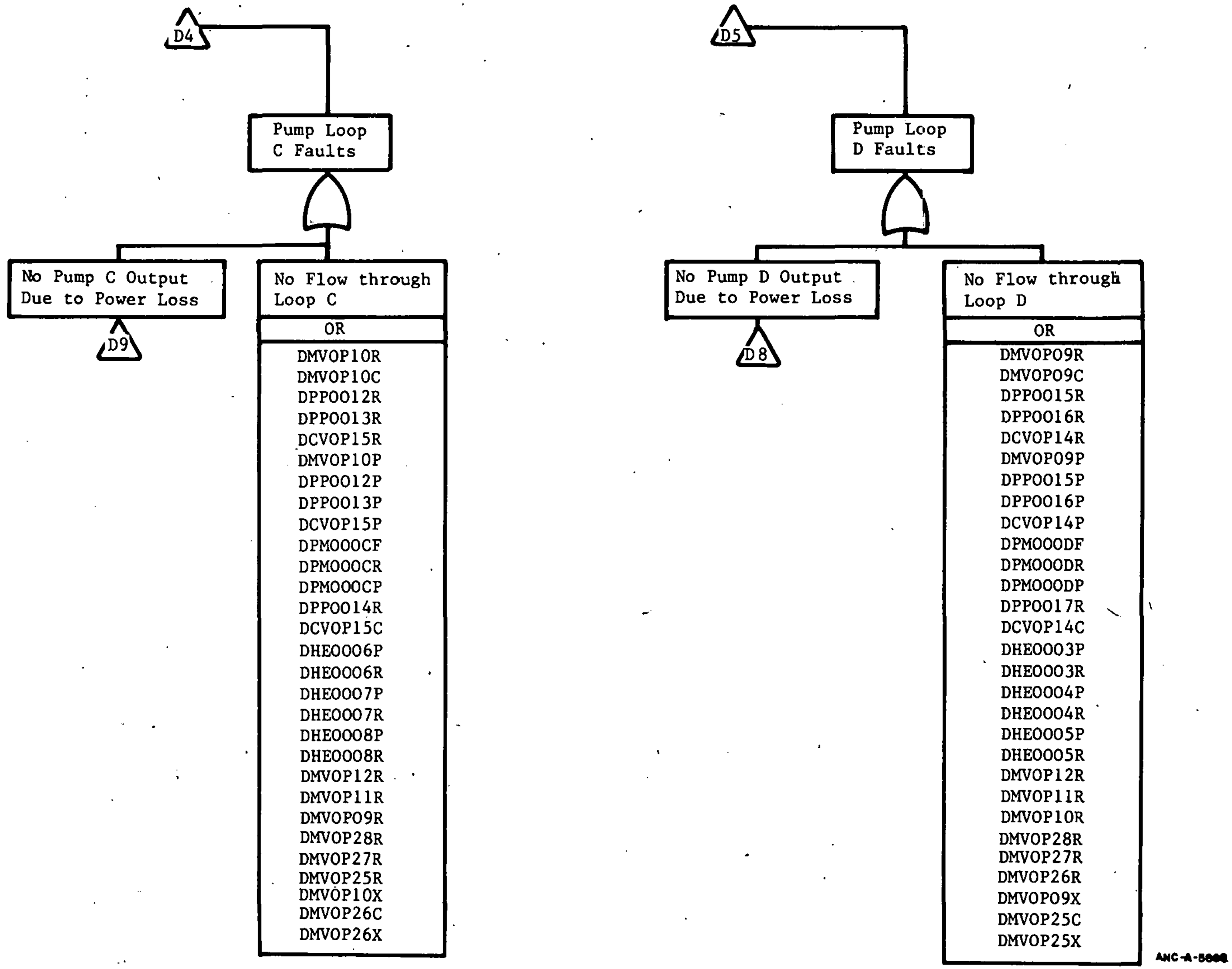

Fig. A-5 ETR loss of forced flow (contd.).

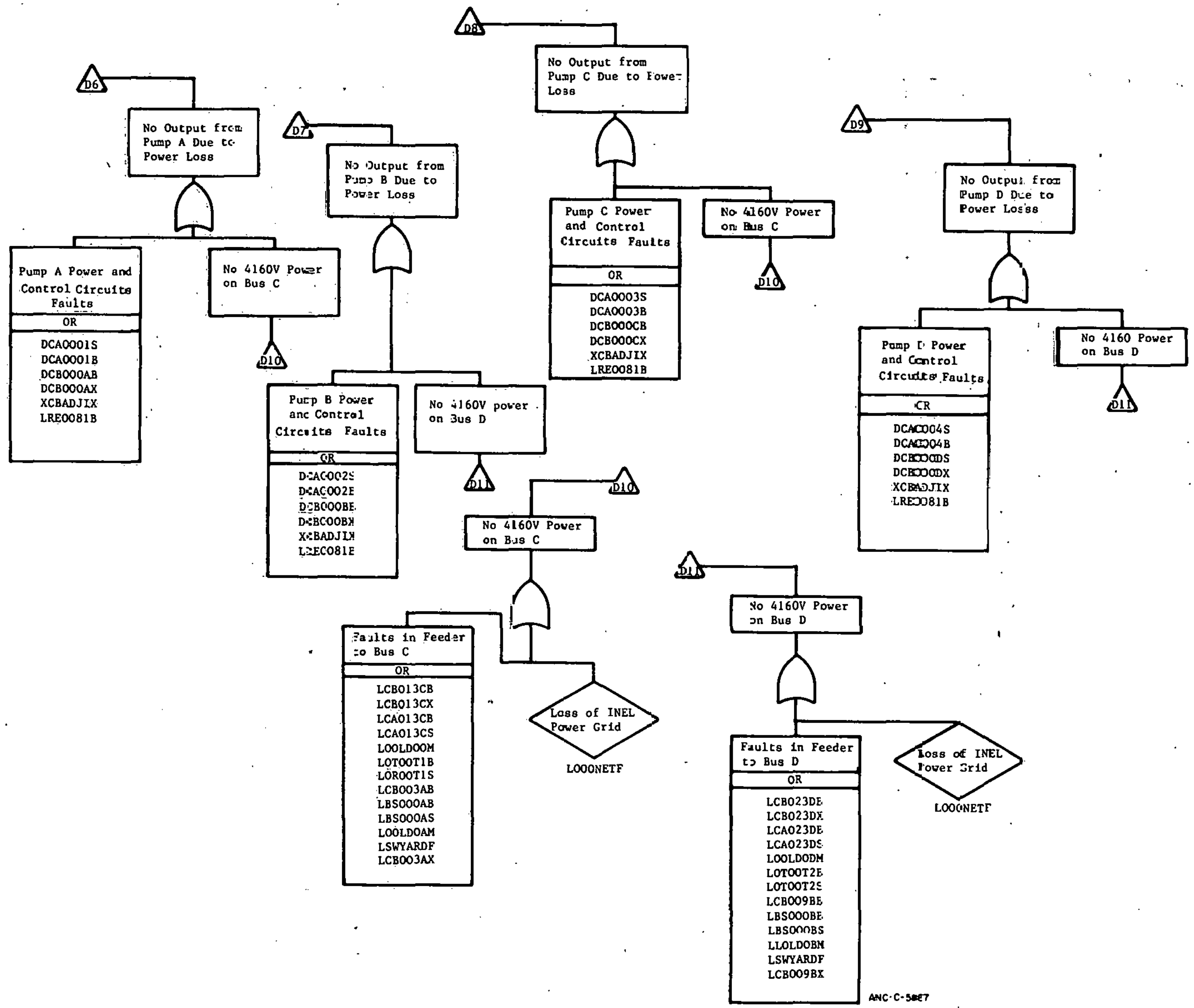

Fig. A-5 ETR loss of forced flow (contd.).

TABLE A-I

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
DPP0001R	PIPE P-1	RUPTURE	10^{-10}	720	30																			PIPE TUNNEL	
DPF0001P	PIPE P-1	PLUGGED	10^{-8}	720	300																			PIPE TUNNEL	
DPF0003R	PIPE P-3	RUPTURE	10^{-10}	720	30																			PIPE TUNNEL	
DXV00P3R	MANUAL VALVE P-3	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DXV00P4R	MANUAL VALVE P-4	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DAVPCV4R	PNEUMATIC CONTROL VALVE PCV4	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DPP0002R	PIPE P-2	RUPTURE	10^{-10}	720	30																			PIPE TUNNEL	
DPP0002P	PIPE P-2	PLUGGED	10^{-10}	720	300																			PIPE TUNNEL	
DXV00P5R	MANUAL VALVE P-5	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DXV00P6R	MANUAL VALVE P-6	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DMV00P8R	BUTTERFLY CONTROL VALVE P-8	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DMV00P8P	BUTTERFLY CONTROL VALVE P-8	PLUGGED	10^{-8}	720	300																			PIPE TUNNEL	
DMV00P8C	BUTTERFLY CONTROL VALVE P-8	FAILS CLOSED	10^{-8}	720	300																			PIPE TUNNEL	
DPP0008R	PIPE P-8	RUPTURE	10^{-10}	720	10																			HEAT EXCHANGE BLDG.	
DPP0008P	PIPE P-8	PLUGGED	10^{-10}	720	300																			HEAT EXCHANGE BLDG.	
DMV0P11R	MOTOR OPERATED VALVE P11	RUPTURE	10^{-8}	720	100	-6 -6																		COMPRESSOR BLDG.	
DMV0P11C	MOTOR OPERATED VALVE P11	CLOSES	10^{-4}	720	30	-6 -1																		COMPRESSOR BLDG.	
DPP0009R	PIPE P-9	RUPTURE	10^{10}	720	30										-5 -2									PUMP CUBICLE	
DPP0010R	PIPE P-10	RUPTURE	10^{-10}	720	30										-5 -2									PUMP CUBICLE	
DCV0P16R	CHECK VALVE P16	RUPTURE	10^{-8}	720	100										-5 -2									PUMP CUBICLE	
DMV0P11P	MOTOR OPERATED VALVE P-11	PLUGGED	10^{-7}	1	300																			PIPE TUNNEL	
DPP0009P	PIPE P-9	PLUGGED	10^{-7}	1	300										-5 -2									PUMP CUBICLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^-x)*																		LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT.	VIBRATION	SOLVENT			
DPP0010P	PIPE P-10	PLUGGED	10^{-7}	1	300										5 / 2									PUMP CUBICLE	
DCV0P16P	CHECK VALVE P16	RUPTURE	10^{-8}	720	100										5 / 2									PUMP CUBICLE	
DPM00BF	PUMP B	NO OUTPUT	10^{-5}	720	10										5 / 2									PUMP CUBICLE	
DPM000BR	PUMP B	RUPTURE	10^{-8}	720	100										5 / 2									PUMP CUBICLE	
DPM000BP	PUMP B	PLUGGED	10^{-6}	1	100										5 / 2									PUMP CUBICLE	
DPP0011R	PIPE 11	RUPTURE	10^{-10}	720	30																			HEAT EXCHANGE BLDG.	
DCV0P29R	CHECK VALVE P29	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DCV0P29C	CHECK VALVE P29	CLOSES	10^{-7}	720	30																			PIPE TUNNEL	
DPP0011P	PIPE P11	PLUGGED	10^{-10}	1	300																			HEAT EXCHANGE BLDG.	
DPP0011R	PIPE P12	RUPTURE	10^{-10}	720	10																			HEAT EXCHANGE BLDG.	
DCV0P30R	CHECK VALVE P30	RUPTURE	10^{-8}	720	10																			HEAT EXCHANGE BLDG.	
DCV0P30C	CHECK VALVE P30	CLOSES	10^{-7}	720	30																			PIPE TUNNEL	
DCV0P17C	CHECK VALVE P17	CLOSES	10^{-7}	720	30										5 / 2									PUMP CUBICLE	
DPM000BF	PUMP B	INSUFFICIENT OUTPUT	10^{-5}	720	10										5 / 2									PUMP CUBICLE	
DPM000BR	PUMP B	RUPTURE	10^{-8}	720	100										5 / 2									PUMP CUBICLE	
DPM000BP	PUMP B	PLUGGED	10^{-6}	1	100										5 / 2									PUMP CUBICLE	
DHE0012P	HEAT EXCHANGED 12	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0012R	HEAT EXCHANGED 12	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0013P	HEAT EXCHANGED 13	PLUGGED	10^{-6}	1	300																			HEAT EXCHANGE BLDG.	
DHE0013R	HEAT EXCHANGED 13	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0014P	HEAT EXCHANGED 14	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0014R	HEAT EXCHANGER 14	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^{-x})*																		LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT			
DPP0013P	PIPE 13	PLUGGED	10^{-7}	1	300										5 2									PUMP CUBICLE	
DCVOP15P	CHECK VALVE 15	PLUGGED	10^{-7}	1	300										5 2									PUMP CUBICLE	
DPM000CF	PUMP C	NO OUTPUT	10^{-5}	720	10										5 2									PUMP CUBICLE	
DPM000CR	PUMP C	RUPTURE	10^{-8}	720	100										5 2									PUMP CUBICLE	
DPM000CP	PUMP C	PLUGGED	10^{-6}	1	100										5 2									PUMP CUBICLE	
DPP0014R	PIPE 14	RUPTURE	10^{-10}	720	30										5 2									PUMP CUBICLE	
DCVOP15C	CHECK VALVE 15	CLOSES	10^{-7}	1	300										5 2									PUMP CUBICLE	
DHE0006P	HEAT EXCHANGER	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0006R	HEAT EXCHANGER	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0007P	HEAT EXCHANGER	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0007R	HEAT EXCHANGER	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0008P	HEAT EXCHANGER	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0008R	HEAT EXCHANGER	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DMVOP12R	MOTOR OPERATED VALVE P12	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP11R	MOTOR OPERATED VALVE P11	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DCVOP16C	CHECK VALVE P16	CLOSES	10^{-7}	1	300										5 2									PUMP CUBICLE	
DHE0009P	HEAT EXCHANGER 9	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0009R	HEAT EXCHANGER 9	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0010P	HEAT EXCHANGER 10	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0010R	HEAT EXCHANGER 10	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0011P	HEAT EXCHANGER 11	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0011R	HEAT EXCHANGER 11	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)

†PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT.	VIBRATION	SOLVENT	LOCATION	REMARKS
DMVOP12R	MOTOR OPERATED VALVE 12	RUPTURE	10^{-8}	720	100																		COMPRESSOR BLDG.	
DMVOP10R	MOTOR OPERATED VALVE P10	RUPTURE	10^{-8}	720	100																		COMPRESSOR BLDG.	
DMVOP10C	MOTOR OPERATED VALVE P10	CLOSED	10^{-4}	720	30																		COMPRESSOR BLDG.	
DPPO012R	PIPE 12	RUPTURE	10^{-10}	720	10										-5 / -2								PUMP CUBICLE	
DPPO013R	PIPE 13	RUPTURE	10^{-10}	720	10										-5 / -2								PUMP CUBICLE	
DCVOP15R	PIPE 15	RUPTURE	10^{-10}	720	10																		HEAT EXCHANGE BLDG.	
DMVOP10P	MOTOR OPERATED VALVE P10	PLUGGED	10^{-7}	1	300																		COMPRESSOR BLDG.	
DPPO012P	PIPE 12	PLUGGED	10^{-7}	1	300																		PUMP CUBICLE	
DPPO009R	PIPE P-9	RUPTURE	10^{-10}	720	30										-5 / -2								PUMP CUBICLE	
DPPO009P	PIPE P-9	PLUGGED	10^{-8}	720	300										-5 / -2								PUMP CUBICLE	
DMVOP12R	MOTOR OPERATED VALVE P-12	RUPTURED	10^{-8}	720	100																		COMPRESSOR BLDG.	
DMVOP12C	MOTOR OPERATED VALVE P-12	CLOSES	10^{-4}	720	30																		COMPRESSOR BLDG.	
DPPO006R	PIPE P-6	RUPTURE	10^{-10}	720	30																		HEAT EXCHANGE BLDG.	
DPPO007R	PIPE P-7	RUPTURE	10^{-10}	720	30																		HEAT EXCHANGE BLDG.	
DCVOP17R	CHECK VALVE P-17	RUPTURE	10^{-8}	720	100										-5 / -2								PUMP CUBICLE	
DMVOP12P	MOTOR OPERATED VALVE P-12	PLUGGED	10^{-7}	1	300																		COMPRESSOR BLDG.	
DMVOP12R	MOTOR OPERATED VALVE P-12	RUPTURE	10^{-8}	720	100																		COMPRESSOR BLDG.	
DPPO006P	PIPE P-6	PLUGGED	10^{-7}	720	300										-5 / -2								PUMP CUBICLE	
DPPO007P	PIPE P-7	PLUGGED	10^{-7}	720	300																		HEAT EXCHANGE BLDG.	
DCVOP17P	CHECK VALVE P-17	PLUGGED	10^{-7}	1	300										-5 / -2								PUMP CUBICLE	
DPMOOOAR	PUMP A	INSUFFICIENT OUTPUT	10^{-5}	720	10										-5 / -2								PUMP CUBICLE	
DPMOOOAR	PUMP A	RUPTURE	10^{-8}	720	100										-5 / -2								PUMP CUBICLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	PRIMARY FAILURE: FAULT DURATION	PRIMARY FAILURE: ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
DMVOP09R	MOTOR OPERATED VALVE P9	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP28P	MOTOR OPERATED VALVE P28	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP27R	MOTOR OPERATED VALVE P27	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP25R	MOTOR OPERATED VALVE P25	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP09R	MOTOR OPERATED VALVE P-9	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP10X	OPERATOR ERROR MOV P-10	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOE BLDG.	
DMVOP26C	MOTOR OPERATED VALVE P26	CLOSES	10^{-4}	720	30																			PIPE TUNNEL	
DMVOP26X	OPERATOR ERROR VALVE P26	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOE BLDG.	
DPP0015R	PIPE 15	RUPTURE	10^{-10}	720	30																			PIPE TUNNEL	
DPP0016K	PIPE 16	RUPTURE	10^{-10}	720	30										-5 -2									PUMP CUBICLE	
DCVOP14R	CHECK VALVE P14	RUPTURE	10^{-8}	720	100										-5 -2									PUMP CUBICLE	
DMVOP09P	MOTOR OPERATED VALVE P9	PLUGGED	10^{-7}	1	300																			PIPE TUNNEL	
DPP0015P	PIPE 15	PLUGGED	10^{-7}	720	100																			PIPE TUNNEL	
DPP0016P	PIPE 16	PLUGGED	10^{-7}	720	100										-5 -2									PUMP CUBICLE	
DCVOP14P	CHECK VALVE P14	PLUGGED	10^{-7}	1	300										-5 -2									PUMP CUBICLE	
DPM000DF	PUMP D	NO FLOW	10^{-5}	720	10										-5 -2									PUMP CUBICLE	
DPM000DP	PUMP D	RUPTURE	10^{-8}	720	100										-5 -2									PUMP CUBICLE	
DPM000DP	PUMP D	PLUGGED	10^{-6}	1	100										-5 -2									PUMP CUBICLE	
DPP0017R	PIPE 17	RUPTURE	10^{-10}	720	30																			HEAT EXCHANGE BLDG.	
DCVOP14C	CHECK VALVE P14	CLOSES	10^{-7}	720	30										-5 -2									PUMP CUBICLE	
DHE003P	HEAT EXCHANGER	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE003R	HEAT EXCHANGER	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
DHE0004P	HEAT EXCHANGER	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0004R	HEAT EXCHANGER	RUPTURED	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DHE0005P	HEAT EXCHANGER	PLUGGED	10^{-8}	1	300																			HEAT EXCHANGE BLDG.	
DHE0005R	HEAT EXCHANGER	RUPTURE	10^{-6}	720	100																			HEAT EXCHANGE BLDG.	
DMVOP12R	MOTOR OPERATED VALVE P12	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP11R	MOTOR OPERATED VALVE P11	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP10R	MOTOR OPERATED VALVE P10	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP28R	MOTOR OPERATED VALVE P28	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP27R	MOTOR OPERATED VALVE P27	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP26R	MOTOR OPERATED VALVE P26	RUPTURE	10^{-8}	720	100																			PIPE TUNNEL	
DMVOP09X	OPERATOR ERROR MOV-9	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOR BLDG.	
DMVOP25C	MOTOR OPERATED VALVE P25	CLOSES	10^{-4}	720	30																			PIPE TUNNEL	
DMVOP25X	OPERATOR ERROR MOV-25	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOR BLDG.	
DMVOP12X	OPERATOR ERROR MOV P-12	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOR BLDG.	
DMVOP28C	MOTOR OPERATED VALVE P-28	CLOSES	10^{-4}	720	30																			PIPE TUNNEL	
DMVOP28X	OPERATOR ERROR MOV P-28	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOR BLDG.	
DMVOP11X	OPERATOR ERROR MOV P-11	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOR BLDG.	
DMVOP27C	MOTOR OPERATED VALVE P27	CLOSES	10^{-4}	720	30																			PIPE TUNNEL	
DMVOP27X	OPERATOR ERROR MOV P27	INADVERTENT CLOSURE	10^{-3}	1	100																			COMPRESSOR BLDG.	
DCA0001S	PUMP A CABLE 1	SHORT	10^{-7}	720	10																			COMPRESSOR BLDG. TO ELEC. BLDG.	
DCA0001B	PUMP A CABLE 1	OPEN	10^{-6}	720	3																			COMPRESSOR TO ELEC. BLDG.	
DCA0002S	PUMP B CABLE 2	SHORT	10^{-7}	720	10																			COMPRESSOR TO ELEC. BLDG.	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT.	VIBRATION	SOLVENT		LOCATION	REMARKS
DCA0002B	PUMP B CABLE 2	OPEN	10^{-6}	720	3	5 1				5 1														COMPRESSOR TO ELEC. BLDG.	
DCA0003S	PUMP C CABLE 3	SHORT	10^{-7}	720	10	5 1				5 1														COMPRESSOR TO ELEC. BLDG.	
DCA0003B	PUMP C CABLE 3	OPEN	10^{-6}	720	3	5 1				5 1														COMPRESSOR TO ELEC. BLDG.	
DCA0004S	PUMP D CABLE 4	SHORT	10^{-7}	720	10	5 1				5 1														COMPRESSOR TO ELEC. BLDG.	
DCA0004B	PUMP D CABLE 4	OPEN	10^{-6}	720	3	5 1				5 1														COMPRESSOR TO ELEC. BLDG.	
DCB0C0AB	PUMP A CIRCUIT BREAKER	OPEN	10^{-6}	720	30	5 1				5 1							1 5							ELEC. BLDG.	
DCBC00AX	OPERATIONAL ERROR CIRCUIT BREAKER	INADVERTENT OPEN	10^{-3}	1	100	5 1				5 1							1 5							ELEC. BLDG.	
XCBADJ1X	OPERATIONAL ERROR CK + BKR ADJUST	INADVERTENT OPEN	10^{-3}	1	100	5 1				5 1							1 5							ELEC. BLDG.	
DCB000BB	PUMP B CIRCUIT BREAKER	OPEN	10^{-6}	720	30	5 1				5 1							1 5							ELEC. BLDG.	
DCB000BX	OPERATIONAL ERROR CIRCUIT BREAKER	INADVERTENT OPEN	10^{-3}	1	100	5 1				5 1							1 5							ELEC. BLDG.	
DCB000CB	PUMP C CIRCUIT BREAKER	OPEN	10^{-6}	720	30	5 1				5 1							1 5							ELEC. BLDG.	
DCB0C0CX	OPERATIONAL ERROR CIRCUIT BREAKER	INADVERTNET OPEN	10^{-3}	1	100	5 1				5 1							1 5							ELEC. BLDG.	
DCB0C0DB	PUMP D CIRCUIT BREAKER	OPEN	10^{-6}	720	30	5 1				5 1							1 5							ELEC. BLDG.	
DCB000DX	OPERATIONAL ERROR CIRCUIT BREAKER D	INADVERTENT OPEN	10^{-3}	1	100	5 1				5 1							1 5							ELEC. BLDG.	
LRE00818	13.8KV FREQUENCY TRIP RELAY	OPEN	10^{-5}	720	100	5 1				5 1							1 5							ELEC. BLDG.	
LCB013CB	BUS C INPUT CIRCUIT BREAKER	OPEN	10^{-6}	720	30	5 1				5 1							1 5							ELEC. BLDG.	
LCB013CX	OPERATIONAL ERROR BUC "C" INCOMING CB	OPEN	10^{-3}	1	100	5 1				5 1							1 5							ELEC. BLDG.	
LCA013CB	INCOMING CABLES TO BUS "C"	OPEN	10^{-6}	720	3	5 1				5 1							1 5							ELEC. BLDG.	
LCA013CS	INCOMING CABLES TC BUS "C"	SHORT	10^{-7}	720	10	5 1				5 1							1 5							ELEC. BLDG.	
LCOLDOCM	OVERLOAD ON BUS C	EXCEED LIMIT	10^{-6}	720	100	5 1				5 1							1 5							ELEC. BLDG.	
LOTOOT1B	TRANSFORMER T-1	OPEN	10^{-6}	720	3	5 1				5 1			1 5			1 5	1 5							TRANSFORMER YARD	
LOTOOT1S	TRANSFORMER T-1	SHORT	10^{-6}	720	3	5 1				5 1			1 5			1 5	1 5							ELEC. BLDG.	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^-x)*																		LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	[illegible]	VIBRATION	SOLVENT			
LCBCO3AB	BUS A CIRCUIT BREAKER 3	OPEN	10^{-6}	720	30	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LCEOO3AX	OPERATIONAL ERROR BUS A CB 3	OPEN	10^{-3}	1	100	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LBSOOOAB	BUS A	OPEN	10^{-6}	720	30	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LBSOOOAS	BUS A	SHORT	10^{-7}	720	30	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LOOLDOAM	OVERLOAD ON BUS A	EXCEEDS LIMIT	10^{-6}	720	100	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LSWYARDF	TEST REACTOR SWITCHYARD FAULTS CAUSE POWER LOSS	LOSS OF FUNCTION	10^{-7}	720	100	-5 / 1				-5 / [illegible]			[illegible] / -5			1 / -5	1 / -5							SWITCHYARD	
LCEO23DB	CIRCUIT BREAKER 23	TRIP OPEN	10^{-6}	720	30	5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LCBO23DX	OPERATIONAL ERROR CIRCUIT BREAKER 23	OPEN	10^{-3}	1	100	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LCAO23DB	CABLE 23	OPEN	10^{-6}	720	3	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LCAO23DS	CABLE 23	SHORT	10^{-7}	720	10	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LOOLDODM	LOAD CIRCUITS ON BUS D	EXCEED LIMIT	10^{-6}	720	100	5 / -1				5 / 1							1 / -5							ELEC. BLDG.	
LOTOOT2B	TRANSFORMER T-2	OPEN	10^{-6}	720	3	-5 / 1				-5 / 1						1 / -5	1 / -5							TRANSFORMER YARD	
LCBOO3BB	CIRCUIT BREAKER 9	TRIP OPEN	10^{-6}	720	30	5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LBSOOOBB	BUS B	OPEN	10^{-6}	720	30	5 / -1				5 / -1							1 / -5							ELEC. BLDG.	
LBSOOOBS	BUS B	SHORT	10^{-7}	720	30	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LOOLDOBM	LOAD CIRCUITS ON BUS B	EXCEED LIMIT	10^{-6}	720	100	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LCBOO9BX	OPERATIONAL ERROR CIRCUIT BREAKER 9	OPEN	10^{-3}	1	100	-5 / -1				-5 / -1							1 / -5							ELEC. BLDG.	
LCONETS	LOSS OF INEL POWER GRID		10^{-5}	1	30	-5 / -1				-5 / -1			1 / -5			1 / -5	1 / -5							INEL	
DXVOP34C	MANUAL VALVE P34	CLOSED	10^{-4}	1	3																			PIPE TUNNEL	
DXVOP34X	OPERATIONAL ERROR MANUAL VALVE P34	INADVERTENT CLOSED	10^{-3}	1	100																			PIPE TUNNEL	
DXVOP34R	MANUAL VALVE P34	RUPTURE	10^{-8}	1	10																			PIPE TUNNEL	
DXVOP34P	MANUAL VALVE P34	PLUGGED	10^{-8}	360	300																			PIPE TUNNEL	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
DPPC020P	PIPE 20	PLUGGED	10^{-8}	360	300																			PIPE TUNNEL	
DPOG20R	PIPE 20	RUPTURE	10^{-10}	360	30																			PIPE TUNNEL	
DORFE14P	ORIFICE FE-4-14	PLUGGED	10^{-8}	1	300																			PIPE TUNNEL	
DXVOP40R	MANUAL VALVE P40	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	
DXVOP41R	MANUAL VALVE P41	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	
DXVOP36R	MANUAL VALVE P36	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	
DXVOP37R	MANUAL VALVE P37	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	
DPP0021R	PIPE 21	RUPTURE	10^{-10}	1	30																			PIPE TUNNEL	
DPP0021P	PIPE 21	PLUGGED	10^{-8}	1	300																			PIPE TUNNEL	
DXV0035P	MANUAL VALVE P35	PLUGGED	10^{-8}	1	300																			PIPE TUNNEL	
DXV0035R	MANUAL VALVE P35	RUPTURE	10^{-8}	1	10																			PIPE TUNNEL	
DXVOP31C	MANUAL VALVE P31	CLOSED	10^{-4}	1	3																			PIPE TUNNEL	
DXVOP31X	OPERATIONAL ERROR MANUAL VALVE P31	CLOSED	10^{-3}	1	100																			PIPE TUNNEL	
DXVOP31R	MANUAL VALVE P31	RUPTURE	10^{-8}	1	10																			PIPE TUNNEL	
DXVOP31P	MANUAL VALVE P31	PLUGGED	10^{-8}	360	300																			PIPE TUNNEL	
DPPC020P	PIPE 20	PLUGGED	10^{-8}	360	300																			PIPE TUNNEL	
DPP0020R	PIPE 20	RUPTURE	10^{-10}	360	30																			PIPE TUNNEL	
DORFE14P	ORIFICE FE-4-14	PLUGGED	10^{-8}	1	300																			PIPE TUNNEL	
DXVOP40R	MANUAL VALVE P40	RUPTURE	10^{-8}	1	10										5 2									EMERG. PUMP CUBICLE	
DXVOP41R	MANUAL VALVE P41	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	
DXVOP36R	MANUAL VALVE P36	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	
DXVOP37R	MANUAL VALVE P37	RUPTURE	10^{-8}	1	10										-5 -2									EMERG. PUMP CUBICLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-I (contd.)

LOSS OF PRIMARY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
DPP0021R	PIPE 21	RUPTURE	10^{-10}	1	30																			PIPE TUNNEL	
DPP0021P	PIPE 21	PLUGGED	10^{-8}	1	300																			PIPE TUNNEL	
DXV0035C	MANUAL VALVE P35	CLOSED	10^{-4}	1	3																			PIPE TUNNEL	
GST0001Z	STRUCTURAL FAILURE OF THE FLOW DISTRIBUTOR	STRUCT. FAILURE	10^{-10}	1	300																			REACTOR VESSEL	
GSTC002Z	STRUCTURAL FAILURE OF EXPERIMENTAL PIPING	STRUCT. FAILURE	10^{-10}	1	300																			REACTOR VESSEL	
GST0003Z	STRUCTURAL FAILURE OF THE UPPER CONTROL ROD SUPPORT FRAME	STRUCT. FAILURE	10^{-10}	1	300																			REACTOR VESSEL	
GSTC004P	CORE BLOCKAGE CAUSED BY FUEL PLATE WARPAGE	STRUCT. FAILURE	10^{-10}	1	300																			REACTOR VESSEL	
GST0005Z	STRUCTURAL FAILURE OF THE CONTROL ROD GUIDE TUBE	STRUCT. FAILURE	10^{-10}	1	300																			REACTOR VESSEL	
D0C0001P	CORE BLOCKAGE CAUSED BY DEBRIS FROM THE REACTOR COOLING SYSTEM	BLOCKAGE	10^{-5}	1	100																			PIPE TUNNEL HX. BLDG.	
D000001X	CORE BLOCKAGE CAUSED BY DEBRIS LEFT IN SYSTEM BY OPERATIONS PERSONNEL	BLOCKAGE	10^{-5}	1	100										-5 -2									COMP. BLDG. PUMP CUBICLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

5.1.3 Results. The probability of a loss of flow event occurring in the primary cooling system was determined through use of the SAMPLE computer code. For this analysis, the events on the fault summary dealing with rupture of the primary system were not included. Therefore, the probability given is for an accident involving failure of the coolant to flow, but the coolant will not be lost from the system.

The fault tree for the primary system was reduced to include only the "significant" fault events. The "significant" events are those used in the probability calculation. An event or combination of events having a probability of occurrence of 1×10^{-10} is insignificant when compared with a probability of 1×10^{-3}. Thus, the fault tree was greatly simplified and the computer analysis time reduced.

The SAMPLE code uses Monte Carlo techniques to determine the system unavailability at a given point in time. The time chosen was 720 hours, which is the assumed length of operating time at the ETR. System unavailability at this time is higher than any other time during the operating period. Unavailability increases as time increases, and unavailability is the probability that the stated event exists at the specified time.

The median probability value of a loss of primary flow accident (loss of flow but not loss of coolant) occurring in the ETR, calculated by the SAMPLE computer code, is 2×10^{-2}/operating cycle. The 90% range is 7×10^{-3} to 1×10^{-1}/operating cycle. The median probability value indicates that about twice in 100 operating cycles, the automatic reactor shutdown system and emergency flow system are expected to be called upon because of a loss of flow event at the ETR.

Four faults resulting from human error are the main contributors to the probability. Their code names on the fault summary, Table A-I, are LCB009BX, LCB023DX, LCB003AX, and LCB013CX. These faults result when electrical circuit breakers are inadvertently opened manually. Two breakers affect power to the A and C primary pumps and two affect power to the B and D pumps. Because three primary pumps are required, loss of two pumps is considered failure of the system. However, at least one primary pump will continue running following any one of these four faults. The flow from this one primary pump exceeds the flow from the emergency flow system. This conservative factor has not been taken into account in this study.

5.2 Plant Protection System

The ETR Plant Protection System (PPS) provides for the automatic shutdown of the reactor in the event one or more of the plant or experimental variables exceeds predefined safe limits. In this analysis failure of the PPS to respond to a loss of flow event is the area of concern. The equipment in the protection system includes the devices that sense or measure the process variables; the instruments that modify, amplify, and operate on the signals from these sensors to provide trip signals to logic elements; the logic elements that receive these trip signals from the instrument channels and provide other trip signals to initiate protective action when the instrument channel trip signal is generated; and finally, the actuators that act directly on the reactor control rod and its release mechanism to provide protective action.

5.2.1 System Description. The PPS is a one-out-of-two logic system; that is, it is comprised, as shown in Figure A-6, of two logic trains either of which produces scram action.

Each automatic shutdown logic train accepts up to 20 scram signals monitored by nuclear and process subsystems or channels. Each channel has its own input buffer, an amplitude and frequency noise rejection circuit, and a latch or seal circuit. The latched scram signal is connected to the 20-input logic train which performs an "OR" function so that the existence of any one or more scram signals at either logic train input produces scram action. At its output, the output scram signal from the logic train is transmitted to all ten actuator controllers, one for each actuator. On receipt of a scram signal from the logic train of either or both automatic shutdown channels, the actuator controllers switch off current to rod-holding magnets, thereby releasing the safety rods. The safety rods are released within 23 milliseconds of receipt of a scram signal at any automatic shutdown channel input.

Manual shutdown Channels 1 and 2 can be activated by the reactor operator by push button switches located on the control console. A manual shutdown is accomplished by interrupting the ac power to the actuator controllers. Either manual shutdown channel will cut off the current to the magnets. The safety rods are released within 250 milliseconds after activation occurs.

5.2.2 System Analysis. The plant protective system was evaluated to identify all events related to the PPS system and its associated subsystems that could prevent reactor shutdown in the event of loss of forced flow through the reactor during operation at power. The loss of flow considered was of a magnitude such that partial or complete core melting would eventually result should the PPS fail.

(1) Assumptions. For purposes of this analysis the following assumptions were made and ground rules established:

(a) One of the ten control rods being inserted into the core is sufficient to scram the reactor; thus, failure to scram is defined as failure of all ten safety rods to enter the core upon a loss of flow signal.

(b) Only three of the twenty input signal variables are most responsive to a loss of flow event: reactor differential pressure, reactor inlet/outlet pressure, and emergency flow. Other protective subsystems response, that is, surge tank level, neutron level, fission break monitor, . . . , lag the selected protective subsystems by several seconds

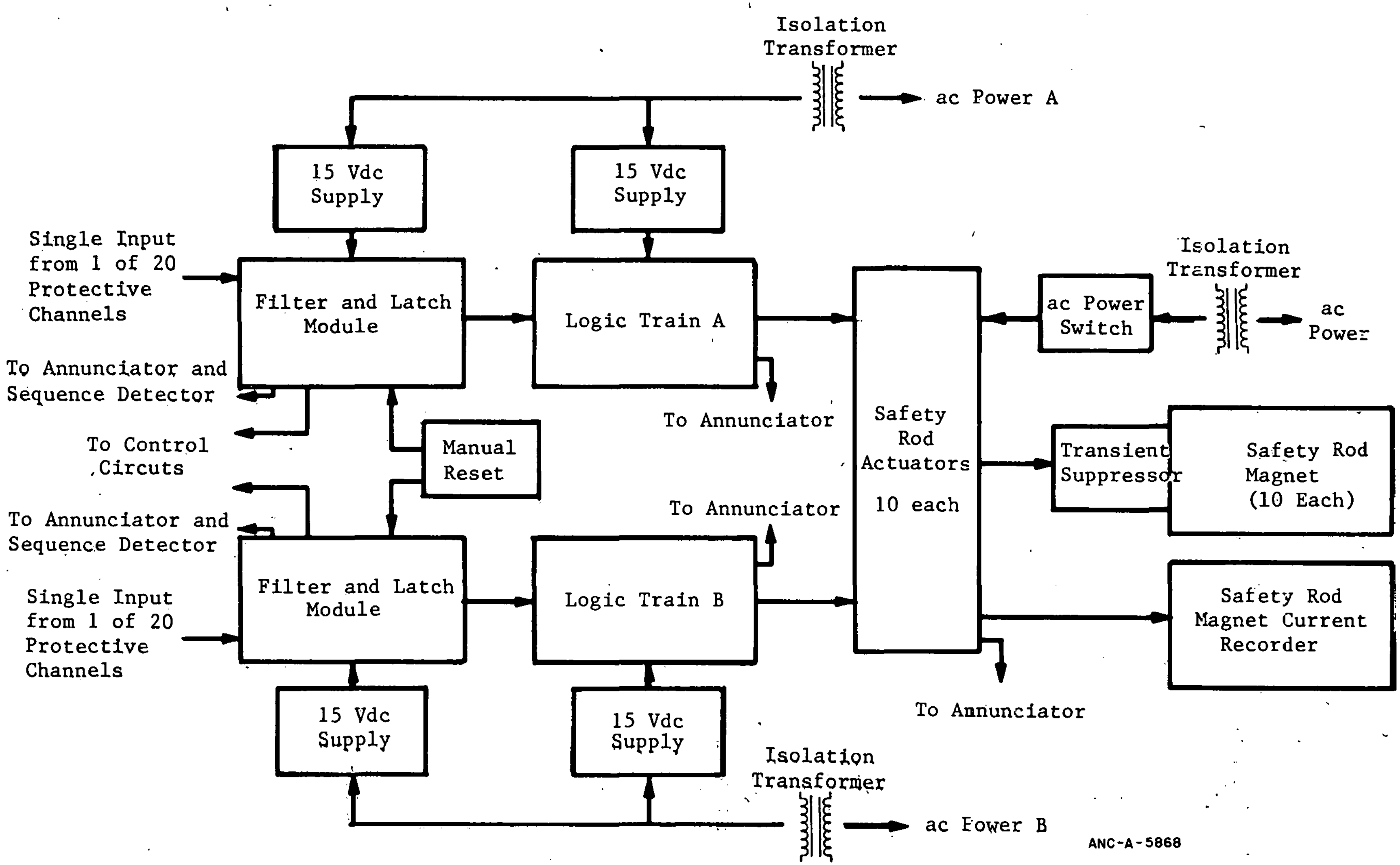

Fig. A-6 Reactor shutdown system simplified block diagram.

(c) Credit can not be given in the analysis for manual operation of the PPS because the operator would not respond quickly enough.

(2) Fault Tree and Fault Summary. The fault tree developed for the PPS response to the loss of flow event and the associated fault summary which lists each fault event are shown in Figure A-7 and Table A-II, respectively. The fault tree on Sheet 1 of Figure A-7 was developed to identify event combinations in Logic Trains A and B which could fail to provide scram signals to the safety rod actuator controllers. No single failures were identified in this analysis; only combinations of two component failures were identified (15 potential fault events in each train which must combine with 15 potential fault events in the other train). These are listed in the fault summary, Table A-II. For the most part, these event combinations are of relatively low probability because no common mode failures exist between events identified in the logic trains.

Sheet 2 of Figure A-7 depicts the fault events within the three required protective channels or subsystems: the differential pressure, the inlet/outlet pressure, and the emergency flow subsystems. The combinations of events identified in the channels needed to fail the PPS are higher order cut sets (higher than the two member cut sets found in the logic train) and, therefore,were not important contributors to system failure.

(3) Data. Data used to quantify the fault model for the PPS were obtained from WASH-1400[A-1] and MIL-HDBK-217B[A-2].

5.2.3 Results. The calculated median value of the unavailability of the PPS as it applies to a loss of flow event is 6×10^{-6}. The upper and lower 90% confidence bounds on this median value are 2×10^{-5} and 2×10^{-6}, respectively.

The principal contributors to this unavailability are transistor faults which could occur in the logic trains. Faults in the individual scram channels which monitor the various scram parameters were insignificant contributors because of their logical arrangement on the fault tree and because of their independence from one another.

5.3 Emergency Flow System

The emergency pumps provide coolant flow through the reactor core in the event of primary coolant pump loss and for circulation of coolant during normal shutdown. In this analysis interest is centered about determining the likelihood that the emergency pumping system will not respond following a loss of forced flow event, that is; interest is centered about the unavailability of the emergency flow system.

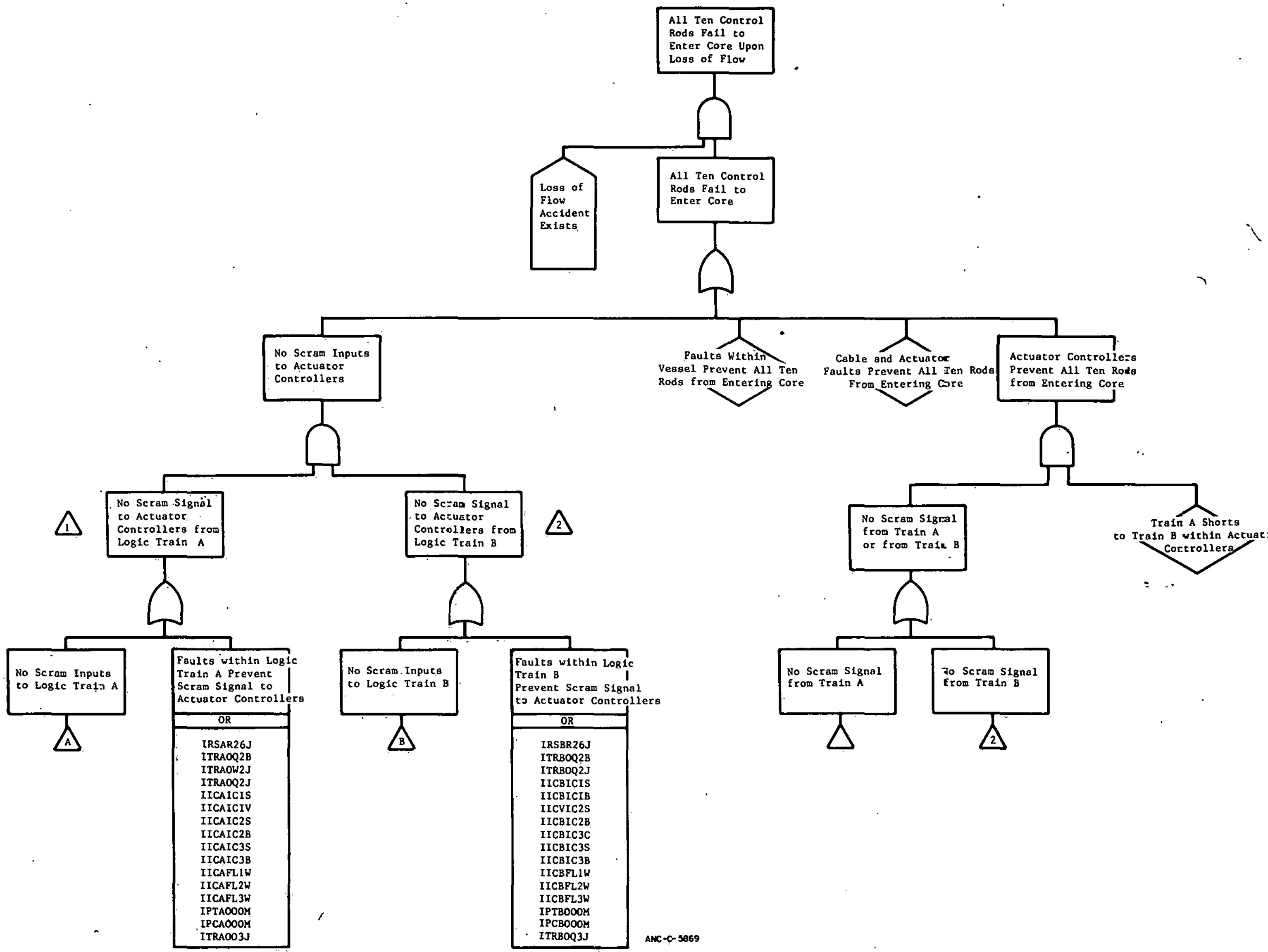

Fig. A-7 Reactor protection system for loss of forced flow fault tree.

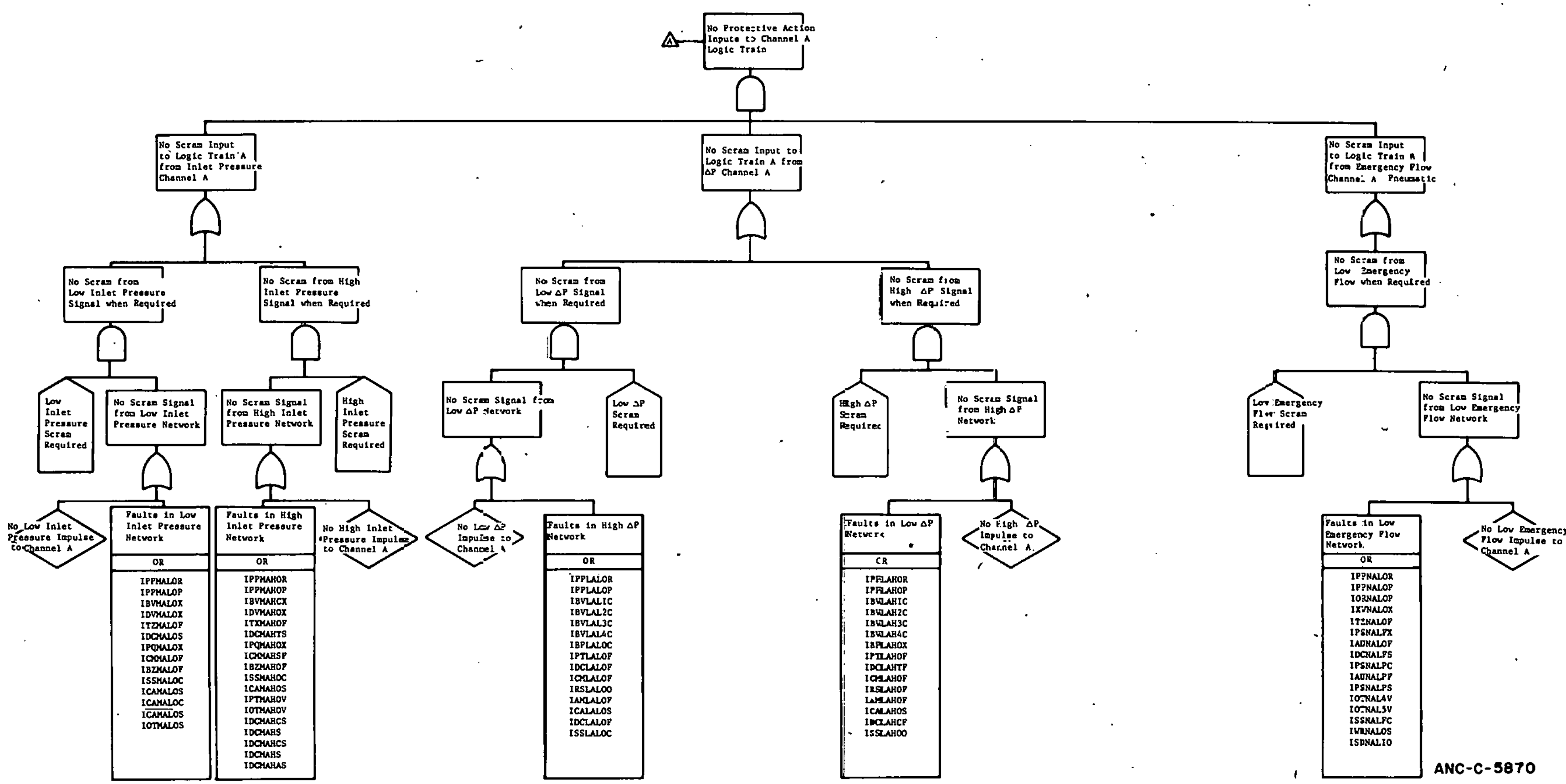

Fig. A-7 Reactor protection system for loss of forced flow fault tree (contd.).

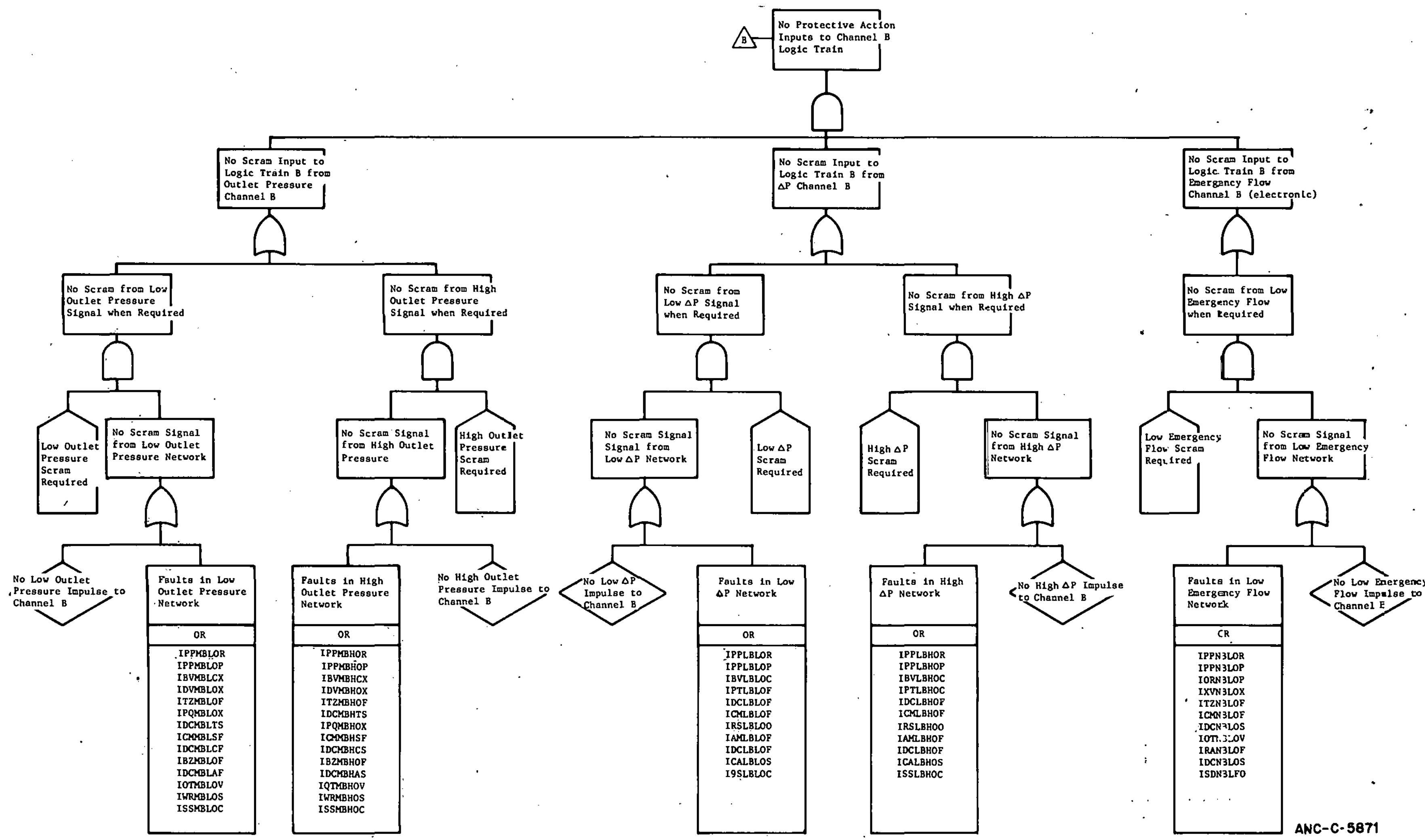

Fig. A-7 Reactor protection system for loss of forced flow fault tree (contd.).

TABLE A-II

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT [LOGIC TRAIN] CHANNEL A AND/OR B	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10⁻¹)*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
ITRAQ93J	LOGIC A TRANSISTOR Q3	SHORT E-C	1×10^{-7}	720	10																			CABINET R-17	
IRSAR26J	LOGIC A RESISTOR R26	SHORT	1×10^{-7}	720	10																			CABINET R-17	
ITRAQ92B	LOGIC A TRANSISTOR Q2	OPEN E-C	1×10^{-7}	720	10																			CABINET R-17	
ITRAQ92J	LOGIC A TRANSISTOR Q2	SHORT B-E	1×10^{-7}	720	10																			CABINET R-17	
IICAIC15	LOGIC A IC1 INTEGRATED CIRCUIT	SHORT TO GROUND	1×10^{-7}	720	10																			CABINET R-17	
IICAIC1B	LOGIC A IC1 INTEGRATED CIRCUIT	OPEN	1×10^{-7}	720	10																			CABINET R-17	
IICAIC25	LOGIC A IC2 INTEGRATED CIRCUIT	SHORT TO GROUND	1×10^{-7}	720	10																			CABINET R-17	
IICAIC2B	LOGIC A IC2 INTEGRATED CIRCUIT	OPEN	1×10^{-7}	720	10																			CABINET R-17	
IICAIC35	LOGIC A IC3 INTEGRATED CIRCUIT	SHORT TO GROUND	1×10^{-7}	720	10																			CABINET R-17	
IICAIC3B	LOGIC A IC3 INTEGRATED CIRCUIT	OPEN	1×10^{-7}	720	10																			CABINET R-17	
IICAFL1W	F & L MODULE (ΔP)	OUTPUT DOES NOT TRANSFER LOW	1×10^{-7}	720	10					10 / 10														CABINET R-17	
IICAFL2W	F & L MODULE (INLET PRESS.	OUTPUT DOES NOT TRANSFER LOW	1×10^{-7}	720	10					10 / 10														CABINET R-17	
IICAFL3W	F & L MODULE (EMER. FLOW	OUTPUT DOES NOT TRANSFER LOW	10^{-7}	720	10					10 / 10														CABINET R-17	
IPTAφOOM	AMSS PHOTOTRANSISTOR	EXCESSIVE CURRENT	10^{-7}	720	10					10 / 10														CABINET R-17	
IPCAφOOM	AMSS PHOTOCOUPLER	EXCESSIVE CURRENT	10^{-7}	720	10					10 / 10														CABINET R-17	
	FAULTS IN LOW ΔP NETWORK CHANNEL #A																								
IPPLALOR	PRESS. IMPULSE LINES	RUPTURE	10^{-7}	720	30																			NORTH PIPE TUNNEL	
IPPLALOP	PRESS. IMPULSE LINES	PLUGGED	10^{-7}	1	30																			NORTH PIPE TUNNEL	
IBVLAL1C	BLOCK VALVE	INADVERTENT CLOSURE	10^{-6}	1	100																			NORTH PIPE TUNNEL	
IBVLAL2C	BLOCK VALVE	INADVERTENT CLOSURE	10^{-6}	1	100																			NORTH PIPE TUNNEL	
IBVLAL3C	BLOCK VALVE	INADVERTENT CLOSURE	10^{-6}	1	100																			NORTH PIPE TUNNEL	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^{-x})*																	LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		
IBVLAL4C	BLOCK VALVE	INADVERTENT CLOSURE	10^{-6}	1	100																		NORTH PIPE TUNNEL	
IBFLALϕX	BY-PASS VALVE	LEFT OPEN	10^{-4}	1	100																		NORTH PIPE TUNNEL	
IPTLALOF	DPT4 - 3A PRESSURE TRANSMITTER	FAILS TO FUNCTION	10^{-5}	720	30																		NORTH PIPE TUNNEL	
IDCLALOF	POWER SUPPLY	LOSS OF FUNCTION	10^{-6}	720	30																		R-29 RSS CAB. NO. CONSOLE	
ICMLALOF	SC4 - 3A SIGNAL COMPARATOR	FAILS TO SCRAM	10^{-5}	720	30																		R-29 RSS CAB. NO. CONSOLE	
IRSLALOϕ	RESISTOR, 200Ω	FAILS OPEN	10^{-7}	720	30																		R-29 RSS CAB. NO. CONSOLE	
IAMLALOF	IA4 - 3A ISOLATION AMPLIFIER	FAILS TO BUFFER	10^{-5}	720	30																		R-29 RSS CAB. NO. CONSOLE	
ICALALOS	INTERCONNECTING CABLES	SHORT CIRCUIT	10^{-7}	720	30																		R-29 RSS CABINET	
IDCLALOF	PSSC4 - 3A S. C. POWER SUPPLY	LOSS OF FUNCTION	10^{-6}	720	30																		R-29 RSS CABINET	
ISSLALOC	FLUX RUN PERMIT SELECTOR SWITCH	RELAY FAILS CLOSED	1.0	/DEMAND	50																		ON CONSOLE	
	FAULTS IN HIGH ΔP NETWORK CHANNEL #A																							
IPPLAHOR	PRESS. IMPULSE LINES	RUPTURE	0.1	720	30																		NORTH PIPE TUNNEL	
IPPLAHOP	PRESS. IMPULSE LINES	PLUGGED	0.1	1	30																		NORTH PIPE TUNNEL	
IBVLAH1C	BV4 - 3A - 1H BLOCK VALVE	CLOSED	1.0	1	100																		NORTH PIPE TUNNEL	
IBVLAH2C	BV4 - 3A - 2H BLOCK VALVE	CLOSED	1.0	1	100																		NORTH PIPE TUNNEL	
IBVLAH3C	BV4 - 3A - 3H BLOCK VALVE	CLOSED	1.0	1	100																		NORTH PIPE TUNNEL	
IBVLAH4C	BV4 - 3A - 4H BLOCK VALVE	CLOSED	1.0	1	100																		NORTH PIPE TUNNEL	
IBPLAHϕX	BPV4 - 3A BY-PASS VALVE (N.C.)	LEFT OPEN	10.0	1	100																		NORTH PIPE TUNNEL	
IPTLAHOF	DPT4 - 3A PRESSURE TRANSMITTER	FAILS TO TRANSMIT HI ΔP SIGNAL	10.0	720	30																		NORTH PIPE TUNNEL	
IDCLAHTF	PST4 - 3A POWER SUPPLY	LOSS OF FUNCTION	1.0	720	30																		INSTRUMENT CABINET CONS. FLOOR	
ICMLAHOF	SC4 - 3A SIGNAL COMPARATOR	FAILS TO SCRAM	10.0	720	30																		INSTRUMENT CABINET NORTH CONS.	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT ($10^{-?}$)*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
IRSLAHOF	RESISTOR, 200Ω	FAILS OPEN	0.1	720	30																			INSTRUMENT CABINET NORTH CONS.	
IAMLAHOF	IA4 - 3A ISOLATION AMPLIFIER	FAILS TO BUFFER	10.0	720	30																			INSTRUMENT CABINET NORTH CONS.	
ICALAHOS	INTERCONNECTING CABLES	SHORT CIRC.	0.1	720	30																			R-29 RSS CABINET	
IDCLAHCF	PSSC 4 - 3A S. C. POWER SUPPLY	LOSS OF FUNCTION	1.0	720	30																			R-29 RSS CABINET	
ISSLAHOФ	FLUX RUN PERMIT SELECTOR SWITCH	RELAY FAILS OPEN	1.0	1/DEMAND	50																			ON CONSOLE	
	[LOW INLET PRESSURE] CHANNEL A																								
IPPMALOR	PRESS. IMPULSE LINES	RUPTURE	0.1	720	30																			PIPE TUNNEL	
IPPMALOP	PRESS. IMPULSE LINES	PLUGGED	0.1	720	30																			PIPE TUNNEL	
IBVMALOX	BLOCK VALVE	INADVERTENT CLOSED	100.0	2	100																			NORTH CONSOLE FLOOR	
IDVMALOX	DRAIN VALVE	INADVERTENT OPEN	100.0	2	100																			NORTH CONSOLE FLOOR	
ITZMALOF	PT4 - 1A PRESS. TRANSMITTER	FAILS TO FUNCTION	10.0	720	30																			NORTH P. TUNNEL WALL	
IDCMALOS	PTV4 - 1A POWER SUPPLY	SHORT TO POWER	1.0	720	30																			SOUTH RSS CABINET	
IPQMALOX	PQ4 - 1A SIGNAL CONDITIONER	IMPROPER CAIN SETTING	100.0	1	100																			SOUTH RSS CABINET	
ICMMALOF	PAS4 - 1A SIGNAL COMPARATOR	FAILS TO FUNCTION	10.0	720	30																			SOUTH RSS CABINET	
IBZMALOF	BZ - P4 - 1A BUFFER AMPLIFIER	FAILS TO FUNCTION	10.0	720	30																			SOUTH RSS CABINET	
ISSMALOC	F. R. PERMIT SWITCH	RELAYS FAIL CLOSED	100.0	/DEMAND	30						-7 -5													ON CONSOLE	
ICAMALOS	INTERCONNECTING CABLES	SHORT CIRCUIT	1.0	720	30																			CONSOLE INSTRUM CABINET	
IOTMALOV	IY - P4 - 1A ISO. TRANSFORMER	DC VOLTAGE SHIFT	1.0	720	30																			NORTH CONSOLE FLOOR	
	[HIGH INLET PRESSURE] CHANNEL A																								
IPPMAHOR	PRESS. IMPULSE LINES	RUPTURE	0.1	720	30																			PIPE TUNNEL	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10⁻ˣ)*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
IPPMAHOP	PRESS. IMPULSE LINES	PLUGGED	0.1	720	30																			PIPE TUNNEL	
IBVMAHCX	BLOCK VALVE	INADVERTENT CLOSED	100.0	2	100																			NORTH CONSOLE FLOOR	
IDVMAHΦX	DRAIN VALVE	INADVERTENT OPEN	100.0	2	100																			NORTH CONSOLE FLOOR	
ITZMAHOF	PT - 4 - 1A PRESS. TRANSMITTER	FAILS TO FUNCTION	100.0	720	30																			NORTH CONSOLE RSS CAB.	
IDCMAHTS	PTV - 4 - 1A POWER SUPPLY	ABNORMAL VOLTAGE	1.0	720	30																			NORTH CONSOLE RSS CAB.	
IPQMAHOX	PQ4 - 1A SIGNAL CONDITIONER	IMPROPER GAIN SETTING	100.0	720	100																			NORTH CONSOLE RSS CAB.	
ICMMAHSF	PAS - 4 - 1A SIGNAL COMPARATOR	FAILS TO SCRAM	10.0	720	30																			NORTH CONSOLE RSS CAB.	
IBZMAHOF	BZ - P4 - 1A BUFFER AMPLIFIER	FAILS TO ISOLATE	10.0	720	30																			NORTH CONSOLE RSS CAB.	
ISSMAHOC	FLUX RUN PERMIT SELECTOR SWITCH	RELAYS FAIL CLOSED	100.0	1/DEMAND	50					7 6														ON CONSOLE	
ICAMAHOS	INTERCONNECTING CABLES	SHORT CIRCUIT	1.0	720	30																			CONSOLE INSTRUMENT CABINETS	
IΦTMAHOV	IY - P4 - 1A ISO. TRANSFORMER	DC VOLTAGE SHIFT	1.0	720	30					6 5														NORTH CONSOLE FLOOR	
IDCMAHCS	PSV4 - 1A COMP. POWER SUPPLY	SHORT TO GROUND	1.0	720	30																			NORTH RSS CABINET	
IDCMAHAS	BZV2 - 1 D. C. POWER SUPPLY	SHORT TO GROUND	1.0	720	30																			NORTH RSS CABINET	
	[LOW EMERGENCY FLOW] CHANNEL A																								
IPPNALOR	IMPULSE LINES	RUPTURE	0.1	720	30																			PIPE TUNNEL	
IPPNALOP	IMPULSE LINES	PLUGGED	0.1	720	30																			PIPE TUNNEL	
IORNALOP	FE - 4 - 14 FLOW ORIFICE	PLUGGED	0.1	720	30																			PIPE TUNNEL	
IXVNALOX	VALVE MANIFOLD	INADVERTENT CLOSURE	100	1/DEMAND	100																			PIPE TUNNEL	
ITZNALOF	FT - 4 - 14A FLOW TRANSMITTER	FAILS TO FUNCTION	10	720	30																			PIPE TUNNEL	
IPSNALFX	FS - 4 - 14A LOW FLOW SWITCH	ERRONEOUS SETTINGS	100	1/DEMAND	50																			NORTH RSS (R-29) CABINET	
IAVNALOF	FAS - 4 - 14A LOW FLOW ALARM MODULE	FAILS TO FUNCTION	10	720	30																			NORTH RSS (R-29) CABINET	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10 [illegible])*																	LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		
IDCNALFS	FSU - 4 - 14 A POWER SUPPLY	SHORT TO POWER	1.0	720	30																		NORTH RSS (R-29) CABINET	
IPSNALPC	PS - 4 - 15A INSTRUMENT AIR LOW PRESSURE SWITCH	FAILS CLOSED	10	720	30																		NORTH RSS (R-29) CABINET	
IAUNALPF	PAS - 4 - 15A LOW PRESSURE ALARM MODULE	FAILS TO FUNCTION	1.0	720	30																		NORTH RSS (R-29) CABINET	
IPSNALPS	PSU - 4 - 15A POWER SUPPLY	SHORT TO POWER	1.0	720	30																		NORTH RSS (R-29) CABINET	
IØTNAL4V	FIY - 4 - 14A ISOLATION TRANSFORMER	VOLTAGE SHIFT	1.0	720	30																		NORTH RSS (R-29) CABINET	
IØTNAL5V	FIY - 4 - 15A ISOLATION TRANSFORMER	VOLTAGE SHIFT	1.0	720	30																		NORTH RSS (R-29) CABINET	
ISSNALFC	FLUX RUN SWITCH	RELAYS FAIL CLOSED	10	1/DEMAND	30																		AMPLIFIER ROOM	
IWRNALOS	INTERCONNECTING WIRING	SHORT CIRCUIT	1.0	720	30																		AMPLIFIER ROOM	
ISDNALIØ	R-28 INSTRUMENT AIR SEQUENCE DETECTOR	OPEN	10	720	30																		AMPLIFIER ROOM	
	[LOW EMERGENCY FLOW] CHANNEL B																							
IPPNBLOR	IMPULSE LINES	RUPTURE	0.1	720	30																		PIPE TUNNEL	
IPPNBLOP	IMPULSE LINES	PLUGGED	0.1	720	30																		PIPE TUNNEL	
IORNBLOP	FE - 4 - 14 FLOW ORIFICE	PLUGGED	0.1	720	30																		PIPE TUNNEL	
IXVNBLOX	VALVE MANIFOLD	INADVERTENT CLOSURE	100	1/DEMAND	100																		PIPE TUNNEL	
ITZNBLOF	FT - 4 - 14B FLOW TRANSMITTER	FAILS TO FUNCTION	10	720	30																		PIPE TUNNEL	
ICMNBLOF	FAS - 4 - 14B SIGNAL COMPARATOR	FAILS TO FUNCTION	10	720	30																		SOUTH RSS (R-30) CABINET	
IDCNBLOS	FSU - 4 - 14B POWER SUPPLY	SHORT TO POWER	1.0	720	30																		SOUTH RSS (R-30) CABINET	
IØTNBLOV	FIY - 4 - 4B ISOLATION TRANSFORMER	DC VOLTAGE SHIFT	1.0	720	30																		SOUTH RSS (R-30) CABINET	
IRANBLOF	FIZ - 4 - 14B TRANSMITTER REMOTE AMPLIFIER	FAILS TO FUNCTION	1.0	720	30																		SOUTH RSS (R-30) CABINET	
IDCNBLOS	IZU - 2 POWER SUPPLY	SHORT TO POWER	1.0	720	30																		SOUTH RSS (R-30) CABINET	
ISDNBLFØ	R-28 E.F. SEQUENCE DETECTOR	OPEN	10	720	30																		R-28 AMPLIFIER ROOM	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-4})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
	[LOW OUTLET PRESSURE] CHANNEL B																								
IPPMBLOR	PRESS. IMPULSE LINES	RUPTURE	0.1	720	30																			PIPE TUNNEL	
IPPMBLOP	PRESS. IMPULSE LINES	PLUGGED	0.1	720	30																			PIPE TUNNEL	
IBVMBLCX	BLOCK VALVE	INADVERTENT CLOSURE	100	1	100																			SOUTH CONSOLE FLOOR	
IDVMBLØX	DRAIN VALVE	INADVERTENT OPEN	100	1	100																			SOUTH CONSOLE FLOOR	
ITZMBLOF	PT - 4 - 1B PRESS. TRANSMITTER	FAILS TO FUNCTION	10	720	30																			P TUNNEL SOUTH WALL	
IPQMBLOX	PQ4 - 1B SIGNAL CONDITIONER	IMPROPER GAIN SETTING	1.0	720	30																			SOUTH RSS CABINET	
IDCMBLTS	PTU4 - 1B POWER SUPPLY & CALIB.	SHORT TO GROUND	1.0	720	30																			SOUTH RSS CABINET	
ICMMBLSF	PAS4 - 1B SIGNAL COMPARATOR	FAILS TO SCRAM	10	720	30																			SOUTH RSS CABINET	
IDCMBLCS	PSV4 - 1B COMP. POWER SUPPLY	SHORT TO GROUND	1.0	720	30																			SOUTH RSS CABINET	
IBZMBLOF	BZ - P4 - 1B BUFFER AMPLIFIER	FAILS TO ISOLATE	10	720	30																			SOUTH RSS CABINET	
IDCMBLAF	BZU2 - 1 D. C. POWER SUPPLY	SHORT TO GROUND	1.0	720	30																			SOUTH RSS CABINET	
IOTMBLOV	IY - P4 - 1B ISO. TRANSFORMER	DC VOLTAGE SHIFT	1.0	720	30																			SOUTH RSS CABINET	
IWRMBLOS	INTERCONNECTING CABLES	SHORT CIRCUIT	1.0	720	30																			SOUTH RSS CABINET	
ISSMBLOC	FLUX RUN PERMIT SELECTOR SWITCH	RELAYS FAIL CLOSED	100	1/DEMAND	50																			ON CONSOLE	
	[HIGH OUTLET PRESSURE] CHANNEL B																								
IPPMBHOR	PRESSURE IMPULSE LINES	RUPTURE	0.1	720	30																			PIPE TUNNEL	
IPPMBHOP	PRESSURE IMPULSE LINES	PLUGGED	0.1	720	30																			PIPE TUNNEL	
IBVMBHCX	BLOCK VALVE	INADVERTENT CLOSURE	100.0	1	100																			SOUTH CONSOLE FLOOR	
IDVMBHØX	DRAIN VALVE	INADVERTENT OPENING	100.0	1	100																			SOUTH CONSOLE FLOOR	
ITZMBHOF	PT - 4 - 1B PRESS. TRANSMITTER	FAILS TO FUNCTION	100.0	720	30																			SOUTH CONSOLE FLOOR	
IDCMBHTS	PTU - 4 - 1B POWER SUPPLY & CALIB.	SHORT TO GROUND	1.0	720	30																			SOUTH RSS CABINET	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

			PRIMARY FAILURE			SECONDARY EVENT (10[illegible])*																			
EVENT NAME	EVENT COMPONENT	FAILURE MODE	FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
IPQMBHOX	PQ4 - 1B SIGNAL CONDITIONER	IMPROPER GAIN SETTING	100.0	720	30																			SOUTH RSS CABINET	
ICMMBHSF	PAS4 - 1B SIGNAL COMPARATOR	FAILS TO SCRAM	10.0	720	30																			SOUTH RSS CABINET	
IDCMBHCS	PSU4 - 1B COMP. POWER SUPPLY	SHORT TO GROUND	1.0	720	30																			SOUTH RSS CABINET	
IBZMBHOF	EZ - P4 - 1B BUFFER AMPLIFIER	FAILS TO ISOLATE	10.0	720	30																			SOUTH RSS CABINET	
IICMBHAS	BZU2 - 1 D. C. POWER SUPPLY	SHORT TO GROUND	1.0	720	30																			SOUTH RSS CABINET	
IQTMBHOV	IY - P4 - 1B ISO. TRANSFORMER	DC VOLTAGE SHIFT	1.0	720	30																			SOUTH RSS CABINET	
IWRMBHOS	INTERCONNECTING CABLES	SHORT CIRCUIT	1.0	720	30																			SOUTH RSS CABINET	
ISSMBHOC	FLUX RUN PERMIT SELECTOR SWITCH	RELAY FAILS CLOSED	100.0	1/DEMAND	50																			ON CONSOLE	
	FAULTS IN LOW ΔP NETWORK CHANNEL B																								
IPPLBLOR	PRESS. IMPULSE LINES	RUPTURE	0.1	720	30																			SOUTH PIPE TUNNEL	
IPPLBLOP	PRESS. IMPULSE LINES	PLUGGED	0.[illegible]	720	30																			SOUTH PIPE TUNNEL	
IBVLBLOC	VALVE MANIFOLD	INADVERTENT CLOSURE	100	720	100																			SOUTH PIPE TUNNEL	
IPTLBLOF	DPT4 - 3B PRESSURE TRANSMITTER	FAILS TO FUNCTION	10	720	30																			SOUTH PIPE TUNNEL	
IDCLBLOF	PST4 - 3B POWER SUPPLY, XTR	LOSS OF FUNCTION	1.0	720	30																			R-3C RSS CAB. SO. CONS.	
ICMLBLOF	SC4 - 3B SIGNAL COMPARATOR	FAILS TO FUNCTION	10	720	30																			R-3C RSS CAB. SO. CONS.	
IRSLBLOØ	RESISTOR, 200Ω	FAILS OPEN	0.1	720	30																			R-3C RSS CAB. SO. CONS.	
IAMLBLOF	IA4 - 3B ISOLATION AMPLIFIER	FAILS TO BUFFER	10	720	30																			R-3C RSS CAB. SO. CONS.	
IDCLBLOF	PSSC4 - 3B POWER SUPPLY, S. C.	LOSS OF FUNCTION	1.0	720	30																			R-3C RSS CAB. SO. CONS.	
ICALBLOS	INTERCONNECTING CABLES	SHORT CIRCUIT	0.1	720	30																			R-3C RSS CAB. SO. CONS.	
ISSLBLOC	FLUX RUN PERMIT SELECTOR SWITCH	RELAY FAILS CLOSED	1.0	1/DEMAND	50																			ON CONSOLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-II (contd.)

REACTOR PROTECTIVE SYSTEM FOR LOSS OF FORCED FLOW

EVENT NAME	EVENT COMPONENT FAULTS IN HIGH ΔP NETWORK CHANNEL B	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	FAULT DURATION	ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP.	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT.	VIBRATION	SOLVENT	LOCATION	REMARKS
IPPLBHOR	PRESS. IMPULSE LINES	RUPTURE	0.1	720	30																		SOUTH PIPE TUNNEL	
IPPLBHOP	PRESS. IMPULSE LINES	PLUGGED	0.1	720	30																		SOUTH PIPE TUNNEL	
IBVLBHOC	VALVE MANIFOLD	INADVERTENT CLOSURE	100	720	100																		SOUTH PIPE TUNNEL	
IPTLBHOC	DPT4 - 3B PRESS. TRANSMITTER	FAILS TO FUNCTION	10	720	30																		SOUTH PIPE TUNNEL	
IDCLBHOF	PST4 - 3B POWER SUPPLY, XTR	LOSS OF FUNCTION	1.0	720	30																		R-30 RSS CAB. SO. CONS.	
ICMLBHOF	SC4 - 3B SIGNAL COMPARATOR	FAILS TO FUNCTION	10	720	30																		R-30 RSS CAB. SO. CONS.	
IRSLBHOϕ	RESISTOR, 200Ω	FAILS OPEN	0.1	720	30																		R-30 RSS CAB. SO. CONS.	
IAMLBHOF	IA4 - 3B ISOLATION AMPLIFIER	FAILS TO BUFFER	10	720	30																		R-30 RSS CAB. SO. CONS.	
IDCLBHOF	PSSC4 - 3B POWER SUPPLY, S. C.	LOSS OF FUNCTION	1.0	720	30																		R-30 RSS CAB. SO. CONS.	
ICALBHOS	INTERCONNECTING CABLES	SHORT CIRCUIT	0.1	720	30																		R-30 RSS CAB. SO. CONS.	
ISSLBHOC	FLUX RUN PERMIT SELECTOR SWITCH	RELAY FAILS CLOSED	1.0	1/DEMAND	50																		ON CONSOLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

5.3.1 System Description. The emergency flow system is comprised of two pumps, their associated controls, the check valves, manual valves, flow element, and piping connected in parallel with the primary coolant Pumps A and B as shown in Figure A-4. The pumps derive power from a battery-backed power source and a diesel power source, both of which supply power during reactor operations. Either pump, by operator selection, can get power from either source, and in the event of loss of any one source, a pump will automatically switch to the other source.

During reactor operations, one pump is normally operating, and the other is on standby. If the operating pump fails, the standby pump is automatically started. Either pump can be designated for operation by manual positioning of selector switches.

Either of the two pumps (2,000 gpm each) is capable of providing adequate coolant circulation for the removal of reactor decay heat.

5.3.2 System Analysis. The emergency flow system was evaluated to determine the probability that either one of the emergency flow pumping systems would not be available when required to respond to a loss of primary coolant flow. Included in the analysis were the instrumentation, controls, and power sources for the pumps.

(1) Assumptions. For purposes of the analysis the following assumptions were made and ground rules established:

(a) Emergency Pump 644-64 is the operating pump, and Pump 644-65 is the standby pump.

(b) If either of the pumps operates, it will provide adequate coolant flow; that is, flow is not degraded.

(c) The primary coolant pumps do not fail in a manner that would block emergency flow in that particular primary loop. The emergency coolant must flow through the primary pumps.

(2) Fault Tree and Fault Summary. Figure A-8 and Table A-III are the fault tree and the fault summary, respectively, developed for the "loss of emergency flow" event. Numerous single events can fail the emergency flow system. For the most part these are relatively low probability events; however, some single events, if they occur, will result in the loss of flow accident and failure of the emergency flow systems. These result from some piping and valves being common to both the primary coolant system and the emergency flow system. Highly potential common mode failures occur in the emergency flow control system because both pumps are controlled from a single control unit, and cables associated with both pumps are routed in the same conduit.

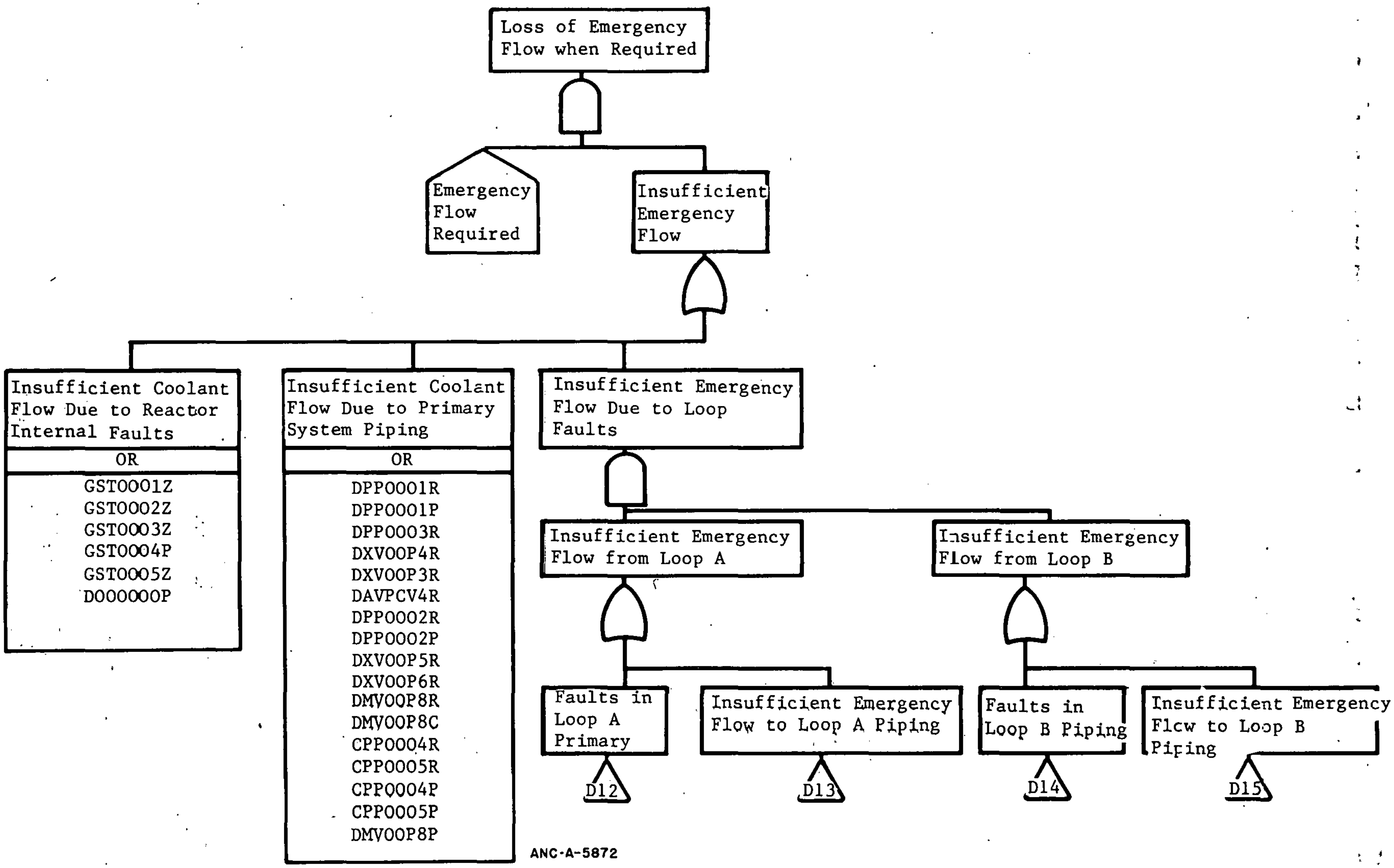

Fig. A-8 Emergency flow system fault tree.

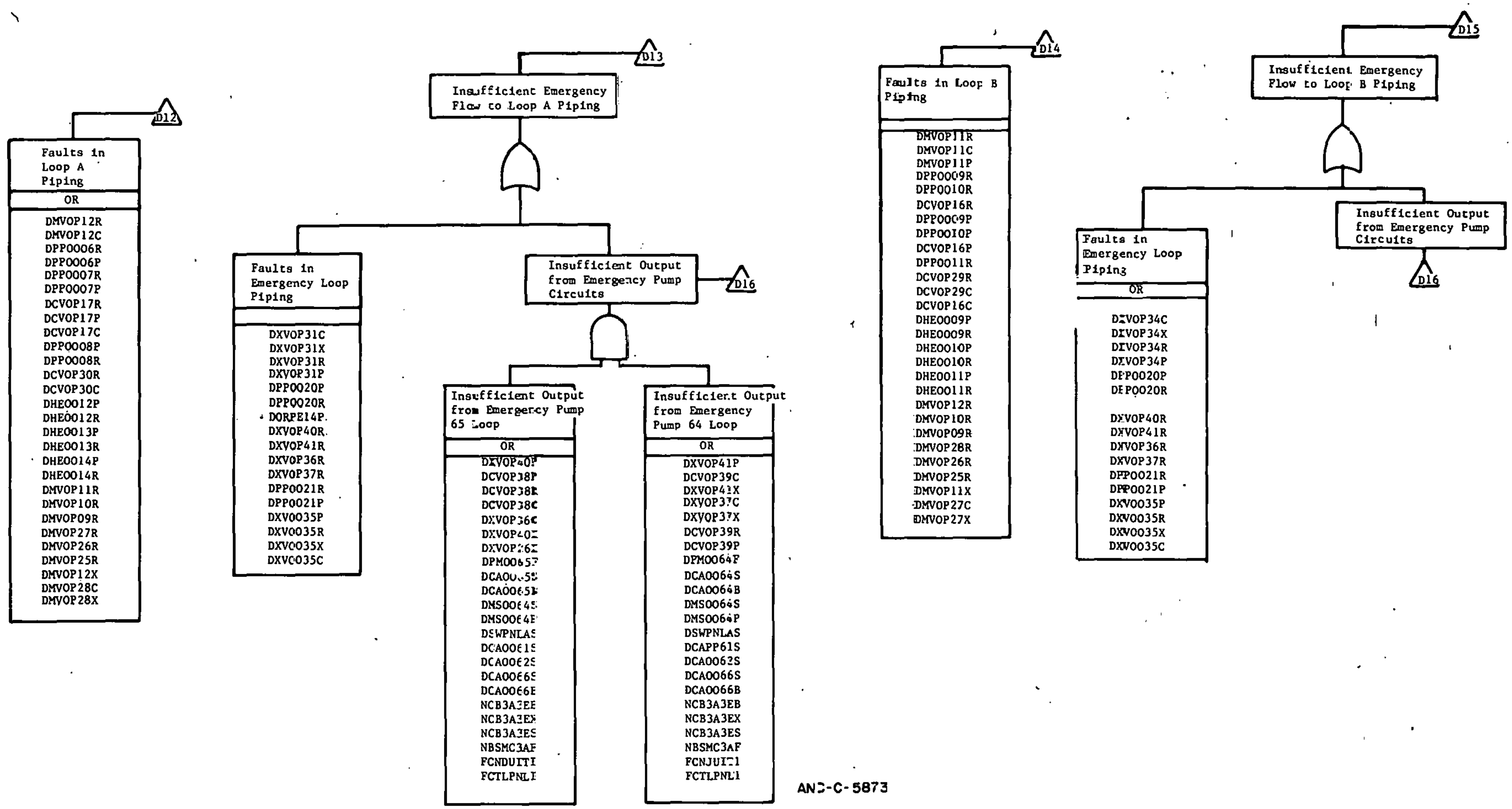

Fig. A-8 Emergency flow system fault tree (contd.).

TABLE A-III

LOSS OF EMERGENCY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^{-x})*																		LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT.	VIBRATION	SOLVENT			
DXVOO35X	OPERATIONAL ERROR MANUAL VALVE P35	INADVERTENT CLOSED	10^{-3}	1	10																			PIPE TUNNEL	
DXVOP40P	MANUAL VALVE P40	PLUGGED	10^{-8}	1	300										-5 -2									EMERG. PUMP CUBICLE	
DCVOP38P	CHECK VALVE P38	PLUGGED	10^{-4}	1	300										-5 -2									EMERG. PUMP CUBICLE	
DCVOP38R	CHECK VALVE P38	RUPTURE	10^{-8}	720	10										-5 -2									EMERG. PUMP CUBICLE	
DCVOP38C	CHECK VALVE P38	CLOSED	10^{-4}	1	30										-5 -2									EMERG. PUMP CUBICLE	
DXVOP36C	MANUAL VALVE P36	CLOSED	10^{-4}	1	3										-5 -2									EMERG. PUMP CUBICLE	
DXVOP40X	OPERATIONAL ERROR VALVE P40	INADVERTENT CLOSED	10^{-3}	1	100										-5 -2									EMERG. PUMP CUBICLE	
DXVOP26X	OPERATIONAL ERROR MANUAL VALVE P36	INADVERTENT CLOSED	10^{-3}	1	100										-5 -2									PIPE TUNNEL	
DPM0064F	EMERG. PUMP 64	NO OUTPUT	10^{-5}	720	10										-5 -2									EMERG. PUMP CUBICLE	
DCA0064S	PUMP 64 POWER CABLE	SHORT TO GRND. (OR PWR)	10^{-7}	720	10	-5 1																		PROCESS C+1 TO PUMP RM.	POWER CABLE FOR BOTH PUMPS IN SAME CONDUIT
DCA0064B	PUMP 64 POWER CABLE	OPEN	10^{-6}	720	10																			PROCESS C+1 TO PUMP RM.	
DMS0064S	EMERG. PUMP CH	SHORTS TO GRND. (OR. PWR.)	10^{-7}	720	10	-5 1																		PROCESS CH ROOM	CONTROLS FOR BOTH PUMPS IN SAME PANEL
DMS0064B	EMERG. PUMP C+1	OPENS	10^{-6}	720	10																			PROCESS C+1 ROOM	
DSWPNLAS	PUMP C TO 1 PNL A	SHORTS	10^{-7}	720	10																			PROCESS C+1 ROOM	
DCA0061S	CONTROL CABLE 61	SHORT TO GRND.	10^{-7}	720	10																			PROCESS C+1 ROOM	
DCA0062S	CONTROL CABLE 62	SHORT TO GRND.	10^{-7}	720	10																			PROCESS C+1 ROOM	
DCA0063S	POWER CABLE 63	SHORT TO GRND.	10^{-7}	720	10	-5 1																		PROCESS C+1 TO ELEC. BLDG	POWER CABLES 63 AND 66 IN SAME CONDUIT
DCA0063B	POWER CABLE 63	OPEN	10^{-6}	720	10																			PROCESS C+1 TO ELEC. BLDG.	
DCA0066S	POWER CABLE 66	SHORT TO GRND.	10^{-7}	720	10	-5 1																		PROCESS CH TO MCC-3A	POWER CABLES 63 AND 66 IN SAME CONDUIT
DCA0066B	POWER CABLE 66	OPEN	10^{-6}	720	10																			PROCESS CH TO MCC-3A	
DCA0065S	PUMP POWER CABLE 65	SHORT TO GRND.	10^{-7}	720	10																			PROCESS CH TO PUMP CUBICLE	
DCA0065B	PUMP POWER CABLE 65	OPEN	10^{-6}	720	10																			PROCESS CH TO PUMP CUBICLE	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-III (contd.)

LOSS OF EMERGENCY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE			SECONDARY EVENT (10^{-x})*																		LOCATION	REMARKS
			FAILURE RATE	FAULT DURATION	ERROR FACTOR	FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT.	VIBRATION	SOLVENT			
NCBPNFFS	DISTRIBUTION PN1 FF CIRCUIT BREAKER	SHORT	10^{-7}	720	10	$\bar{5}$ / $\bar{1}$				$\bar{5}$ / $\bar{1}$							1 / $\bar{5}$							ELEC. BLDG.	
NCBPNFFB	DISTRIBUTION PN1 FF CIRCUIT BREAKER	OPEN	10^{-6}	720	10	$\bar{5}$ / $\bar{1}$				$\bar{5}$ / $\bar{1}$							1 / $\bar{5}$							ELEC. BLDG.	
NCBPNFFX	OPERATIONAL ERROR CK + BKR FF	INADVERTENT OPEN	10^{-3}	1	100	$\bar{5}$ / 1				$\bar{5}$ / $\bar{1}$							1 / $\bar{5}$							ELEC. BLDG.	
NCB3A3ES	MCC-3A COMPT.3E CK + BKR	SHORT	10^{-7}	720	10										$\bar{5}$ / $\bar{2}$									COMPRESSOR BLDG.	
NCB3A3EB	MCC-3A COMPT.3E CK + BKR	OPEN	10^{-6}	720	10										$\bar{5}$ / $\bar{2}$									COMPRESSOR BLDG.	
NCB3A3EX	OPERATIONAL ERROR MCC-3ACD 3E	OPEN	10^{-3}	1	100										$\bar{5}$ / $\bar{2}$									COMPRESSOR BLDG.	
NBSMC3AF	MCC-3A	NO POWER	10^{-3}	720	10										$\bar{5}$ / $\bar{2}$									COMPRESSOR BLDG.	
DXVOP41P	MANUAL VALVE P41	PLUGGED	10^{-8}	1	300										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DCVOP39C	CHECK VALVE P39	CLOSED	10^{-4}	1	300										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DXVOP41X	OPERATIONAL ERROR MANUAL VALVE P41	INADVERTENT CLOSED	10^{-3}	1	100										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DXVOP37C	MANUAL VALVE P37	CLOSED	10^{-4}	1	30										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DXVOP37X	OPERATIONAL ERROR MANUAL VALVE P37	INADVERTENT CLOSED	10^{-3}	1	100										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DCVOP39R	CHECK VALVE P39	RUPTURE	10^{-8}	1	100										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DCVOP39P	CHECK VALVE P39	PLUGGED	10^{-4}	1	100										$\bar{5}$ / $\bar{2}$									EMERG. PUMP CUBICLE	
DPM0065F	EMERGENCY PUMP 65	NO OUTPUT	10^{-5}	720	10										5 / 2									EMERG. PUMP CUBICLE	
NBS0027F	DISTRIBUTION PANEL 348-27	NO POWER	10^{-3}	1	10	5 / 1				5 / 1														ELEC. BLDG.	
FCNDUIT1	FIRE DUE TO SHORT IN CONDUIT #1		10^{-5}	1	10																				
FCTLPNL1	FIRE IN EMERG. PUMP C+1 PN1		10^{-5}	1	10																			PROCESS C+1 ROOM	
DXVOP31R	MANUAL VALVE P31	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DXVOP31P	MANUAL VALVE P31	PLUGGED	10^{-4}	1	300																			PIPE TUNNEL	
DPP0020P	PIPE 20	PLUGGED	10^{-10}	720	300																			PIPE TUNNEL	
DPP0020R	PIPE 20	RUPTURE	10^{-10}	720	300																			PIPE TUNNEL	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

TABLE A-III (contd.)

LOSS OF EMERGENCY FLOW

EVENT NAME	EVENT COMPONENT	FAILURE MODE	PRIMARY FAILURE: FAILURE RATE	PRIMARY FAILURE: FAULT DURATION	PRIMARY FAILURE: ERROR FACTOR	SECONDARY EVENT (10^{-x})*: FIRE	FLOOD	EARTHQUAKE	STEAM	HI-TEMP	PRESSURE	HUMIDITY	FREEZING	MISSILE	PIPE WHIP	WIND	DUST	RADIATION	CORROSION	INDUCED VOLT	VIBRATION	SOLVENT		LOCATION	REMARKS
DØRFE14P	ORIFICE FE-14	PLUGGED	10^{-5}	1	100										-5 -2									PUMP CUBICLE	
DXVOP40R	MANUAL VALVE P40	RUPTURE	10^{-8}	720	10										-5 -2									PUMP CUBICLE	
DXVOP41R	MANUAL VALVE P41	RUPTURE	10^{-8}	720	10										-5 -2									PUMP CUBICLE	
DXVOP36R	MANUAL VALVE P36	RUPTURE	10^{-8}	720	10										-5 -2									PUMP CUBICLE	
DXVOP37R	MANUAL VALVE P37	RUPTURE	10^{-8}	720	10										-5 -2									PUMP CUBICLE	
DPPOO21R	PIPE 21	RUPTURE	10^{-10}	720	300																			PIPE TUNNEL	
DPPOO21P	PIPE 21	PLUGGED	10^{-10}	720	300																			PIPE TUNNEL	
DXVCO35P	MANUAL VALVE P35	PLUGGED	10^{-4}	1	300																			PIPE TUNNEL	
DXVCO35R	MANUAL VALVE P35	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DXVOP34R	MANUAL VALVE P34	RUPTURE	10^{-8}	720	10																			PIPE TUNNEL	
DXVOP34P	MANUAL VALVE P34	PLUGGED	10^{-4}	1	300																			PIPE TUNNEL	

*PROBABILITY OF SECONDARY EVENT OCCURRING (UPPER NUMBER)
PROBABILITY OF COMPONENT FAILURE GIVEN SECONDARY EVENT (LOWER NUMBER)

(3) Data. Data used to quantify the emergency flow system fault model were obtained from Reference A-1.

5.3.3 Results. The emergency flow system was analyzed using the SAMPLE computer code in the same manner explained in Section 5.1.3. The result for the median probability value for failure of the emergency flow system to respond to a loss of flow accident is 6×10^{-3}/operating cycle. The 90% range is from 2×10^{-3} to 6×10^{-2}/operating cycle.

The probability reported is the probability that the emergency flow system is not available when called upon following a loss of primary flow accident. The probability that the system will fail to continue operating after accident initiation is assumed to be unity or that it will not fail after emergency flow is established. The significant contributors to the reported probability were:

(1) Closure or plugging of P35 manual valve.

(2) Fire in Conduit 1 or emergency pump panel.

(3) Open motor starter to emergency pumps.

(4) Shorts in emergency circuits.

6. EVENT SEQUENCE PROBABILITIES

The probability of occurrence of the three event sequences on the event tree, Figure A-3, was found using the SAMPLE code. The code was used to calculate probabilities of failure for combined fault trees. Therefore, any dependence between systems is taken into account.

The probability of success is equal to one minus the probability of failure. For the automatic safety system with a probability of failure of 6.2×10^{-6}, the probability of success would be 0.9999938 or very close to unity. The probability of success used in all cases was, therefore, unity.

The following symbols are used in the calculations for probabilities.

	Success	Failure
Loss of Primary Flow		$P_{\bar{F}}$
Automatic Reactor Shutdown System	P_S	$P_{\bar{S}}$
Emergency Flow System	P_E	$P_{\bar{E}}$

Sequence one is the sequence of events that should happen following a loss of primary flow accident. The automatic reactor shutdown system and the emergency flow system function properly. The probability of S_1 occurring is equal to

$$P_{\bar{F}} \times P_S \times P_E = P_{\bar{F}} \times 1 \times 1 = P_{\bar{F}}$$

as reported in Section 5.1.3, the median probability value of $P_{\bar{F}}$ is 2×10^{-2}/operating cycle, and the 90% range is 7×10^{-3} to 1×10^{-1}/operating cycle.

Sequence two is the sequence in which the automatic safety system functions properly but the emergency flow system fails. To find this probability, the fault trees for the loss of primary flow accident and failure of the emergency flow system were combined. The combined trees were analyzed with the SAMPLE code and the median probability value is 8×10^{-4}/operating cycle. The 90% range is 1×10^{-4} to 2×10^{-2}/operating cycle. This probability cannot be found by

$$P = P_{\bar{F}} \times P_S \times P_{\bar{E}}$$

because the emergency flow system is not independent of the primary system. Sequence three results if the automatic reactor shutdown system fails. The emergency flow system is not included because core melting will occur regardless of the emergency flow system action. Manual actuation of the shutdown system is available to prevent this sequence from occurring, but has not been credited in this study. To determine the probability of S3, the fault trees for loss of primary flow and the automatic safety system were combined and analyzed with the SAMPLE code. Because of complete independence between the two systems, the median value of probability can be found by combining the probability values obtained by evaluating individual trees; that is:

$$P = P_{\bar{F}} \times P_{\bar{S}}$$

but the range cannot be found without combining the trees and using the SAMPLE code. The resultant median probability value is 1×10^{-7}/operating cycle and the range is 3×10^{-8} to 9×10^{-7}.

7. CORE DAMAGE

Once sequence three, (S3) in Figure A-3, is shown to result in core melting, the amount of core damage becomes important. The degree of core damage expected under given conditions can be obtained by use of a computer program designated RELAP4[A-3], written in FORTRAN IV, that describes the behavior of water-cooled nuclear reactors during postulated accidents such as loss of coolant, loss of flow, pump failure, or power transients.

7.1 Model

RELAP4 was used to evaluate the ETR loss of flow accidents (with and without emergency flow and with and without reactor scram).

The reactor core was modeled as 20 volumes connected to upper and lower plenums representing the reactor tank as shown in Figure A-9. The two plenums, Volumes 1 and 2, were input as timed volumes with the inlet pressure held constant and the outlet pressure increased to model the flow coastdown expected when two primary coolant pumps are used. The flow coastdown expected with two pumps operating is shown in Table A-IV.

7.2 Geometric Description

Volumes, flow areas, elevations, and volume heights are necessary geometric input parameters to RELAP4. The volume geometries are described in Tables A-V and A-VI, and the core section geometries are described in Table A-VII. Junction data are listed in Tables A-VIII and A-IX.

7.3 Heat Transfer

A heat conductor model is used to account for heat transfer to and from the fluid in given volumes. The heat addition rate to the fluid is calculated as the product of the surface flux and the heat transfer area at the conductor surface adjacent to the fluid. The bases for the ETR heat transfer correlations are as follows.

7.3.1 Bases. RELAP4 has several correlations which are used in calculating heat transfer rates to the coolant. The correlation used is dependent on the thermodynamics of the coolant. The correlations and the applicable conditions are discussed in the RELAP4 manual[A-3].

The user can choose the correlations to be used to calculate the critical heat flux and subsequent film boiling heat transfer rates. For this analysis, the Savannah River correlation[A-4] was used to predict the critical heat flux, and the Dougall and Rohsenow correlation[A-5] was used to predict the film boiling heat transfer rate.

For the core, a metal-water ratio of four was used to calculate the mass of metal. This mass was incorporated adjacent to the core reflector volumes. The upper and lower plenums used the vessel as a heat sink.

7.3.2 Slab Modeling. In RELAP4 all heat generating or absorbing surfaces adjacent to volumes are termed slabs. Slabs having internal heat generation capabilities (core slabs) are assigned additional properties. Characteristic properties are defined for slabs. Table A-VII defines the volumes adjacent to the slabs and the characteristic properties defined for each slab.

The geometry values listed in Table A-X were derived from the basic data listed on system drawings. The core slab properties were derived from the data in Table A-XI. The derivation of the heat generation values in Table A-XII was based on the power profiles measured in the Engineering Test Reactor Critical (ETRC) Facility for the SLSF core at the beginning of life. Table A-XII shows the makeup of each volume and the power distributions in the elements and corresponding volumes. The vertical profile was established from vertical flux profiles measured in the ETRC.

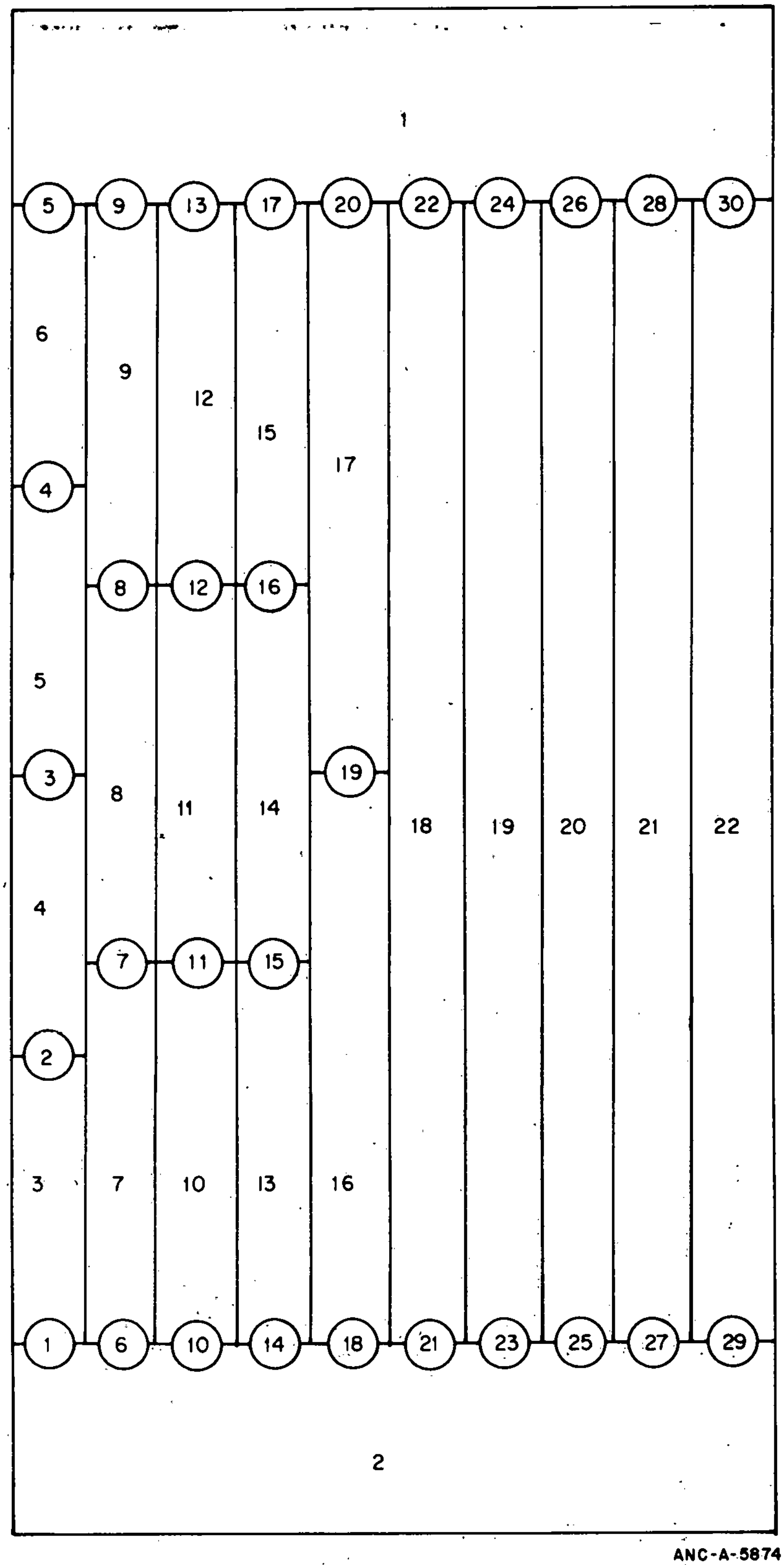

Fig. A-9 ETR core model for risk evaluation.

TABLE A-IV

FLOW COASTDOWN EXPECTED WITH TWO PUMPS OPERATING

TIME DEPENDENT TABLES FOR 2 VOLUMES,

SET NUM	NUM PTS	TIME (SECONDS)	PRESSURE (PSIA)	TEMPERATURE (DEG F)	MIXTURE QUALITY	MIXTURE LEVEL (FEET)	
1		0.	.176500E+03	.137000E+03	0.	.900000E+01	(VOL 2)
	20	.200000E+00	.171000E+03	.137000E+03	0.	.900000E+01	
		.300000E+00	.172200E+03	.137000E+03	0.	.900000E+01	
		.500000E+00	.175500E+03	.137000E+03	0.	.900000E+01	
		.700000E+00	.178600E+03	.137000E+03	0.	.900000E+01	
		.900000E+00	.182000E+03	.137000E+03	0.	.900000E+01	
		.150000E+01	.191000E+03	.137000E+03	0.	.900000E+01	
		.200000E+01	.196000E+03	.137000E+03	0.	.900000E+01	
		.250000E+01	.199600E+03	.137000E+03	0.	.900000E+01	
		.300000E+01	.202500E+03	.137000E+03	0.	.900000E+01	
		.350000E+01	.204500E+03	.137000E+03	0.	.900000E+01	
		.400000E+01	.206000E+03	.137000E+03	0.	.900000E+01	
		.500000E+01	.208400E+03	.137000E+03	0.	.900000E+01	
		.600000E+01	.209700E+03	.137000E+03	0.	.900000E+01	
		.700000E+01	.210700E+03	.137000E+03	0.	.900000E+01	
		.800000E+01	.211200E+03	.137000E+03	0.	.900000E+01	
		.100000E+02	.211800E+03	.137000E+03	0.	.900000E+01	
		.150000E+02	.212400E+03	.137000E+03	0.	.900000E+01	
		.260000E+02	.213400E+03	.137000E+03	0.	.900000E+01	
		.600000E+03	.213400E+03	.137000E+03	0.	.900000E+01	
2		0.	.205500E+03	.110000E+03	0.	.226000E+02	(VOL 1)
	0						

TABLE A-V

CORE VOLUME GEOMETRY

VOLUME DATA ACTUALLY BEING USED.

VOL NUM	2-PH FRIC	FLOW AREA (FT**2)	EQUIVALENT DIAMETER (FT)	LENGTH (FT)	L/2A (FT**-1)	HORIZ. AREA (FT**2)	TEMPERATURE (F)	SATURATION TEMP. (F)
1	0	.102400E+03	.114184E+02	.173262E+02	.846004E-01	.785044E+02	.110000E+03	.384070E+03
2	0	.412800E+02	.724978E+01	.765988E+01	.927796E-01	.551333E+02	.137000E+03	.368647E+03
3	0	.388000E-01	.200000E-01	.670103E+00	.863535E+01	.333333E-01	.183700E+03	.371688E+03
4	0	.388000E-01	.200000E-01	.832474E+00	.107278E+02	.380000E-01	.155800E+03	.374415E+03
5	0	.388000E-01	.200000E-01	.832474E+00	.107278E+02	.380000E-01	.129600E+03	.377079E+03
6	0	.388000E-01	.200000E-01	.111856E+01	.144144E+02	.620000E-01	.112400E+03	.379675E+03
7	0	.194000E+00	.200000E-01	.104124E+01	.268360E+01	.192381E+00	.179000E+03	.372148E+03
8	0	.194000E+00	.200000E-01	.104124E+01	.268360E+01	.192381E+00	.140000E+03	.375757E+03
9	0	.194000E+00	.200000E-01	.104124E+01	.268360E+01	.192381E+00	.115000E+03	.378816E+03
10	0	.155000E+00	.200000E-01	.103871E+01	.335068E+01	.153333E+00	.174600E+03	.372148E+03
11	0	.155000E+00	.200000E-01	.103871E+01	.335068E+01	.153333E+00	.138000E+03	.375757E+03
12	0	.155000E+00	.200000E-01	.103871E+01	.335068E+01	.153333E+00	.114200E+03	.378816E+03
13	0	.233000E+00	.200000E-01	.103863E+01	.222881E+01	.230476E+00	.171400E+03	.372148E+03
14	0	.233000E+00	.200000E-01	.103863E+01	.222881E+01	.230476E+00	.136000E+03	.375757E+03
15	0	.233000E+00	.200000E-01	.103863E+01	.222881E+01	.230476E+00	.113500E+03	.378816E+03
16	0	.310000E+00	.200000E-01	.156129E+01	.251821E+01	.308280E+00	.159100E+03	.373062E+03
17	0	.310000E+00	.200000E-01	.156129E+01	.251821E+01	.310256E+00	.120100E+03	.378385E+03
18	0	.272000E+00	.200000E-01	.311397E+01	.572421E+01	.272347E+00	.133000E+03	.375757E+03
19	0	.272000E+00	.170000E-01	.311397E+01	.572421E+01	.272347E+00	.132000E+03	.375757E+03
20	0	.350000E+00	.170000E-01	.310000E+01	.442857E+01	.348875E+00	.130000E+03	.375757E+03
21	0	.110400E+01	.200000E-01	.312500E+01	.141531E+01	.110932E+01	.115000E+03	.375757E+03
22	0	.110400E+01	.200000E-01	.312500E+01	.141531E+01	.110932E+01	.112000E+03	.375757E+03

TABLE A-VI

CORE VOLUME GEOMETRY

VOLUME DATA ACTUALLY BEING USED.

VOL NUM	BUBL INDX	TIME DEP	PRESSURE (PSIA)	ENTHALPY	VOLUME (FT**3)	HEIGHT (FT)	MIXTURE LEVEL (FT)	ELEVATION (FT)
1	1	2	.205500E+03	.785127E+02	.177420E+04	.226000E+02	.226000E+02	.816000E+02
2	1	1	.170500E+03	.105371E+03	.316200E+03	.900000E+01	.900000E+01	.695700E+02
3	2	0	.177000E+03	.152108E+03	.260000E-01	.780000E+00	.780000E+00	.785200E+02
4	2	0	.185000E+03	.124189E+03	.323000E-01	.850000E+00	.850000E+00	.792700E+02
5	2	0	.189000E+03	.960283E+02	.323000E-01	.850000E+00	.850000E+00	.801000E+02
6	2	0	.195000E+03	.808792E+02	.434000E-01	.700000E+00	.700000E+00	.809300E+02
7	2	0	.178000E+03	.147401E+03	.202000E+00	.105000E+01	.105000E+01	.785200E+02
8	2	0	.186000E+03	.108407E+03	.202000E+00	.105000E+01	.105000E+01	.795500E+02
9	2	0	.193000E+03	.834674E+02	.202000E+00	.105000E+01	.105000E+01	.805800E+02
10	2	0	.178000E+03	.142991E+03	.161000E+00	.105000E+01	.105000E+01	.785200E+02
11	2	0	.186000E+03	.106408E+03	.161000E+00	.105000E+01	.105000E+01	.795500E+02
12	2	0	.193000E+03	.826695E+02	.161000E+00	.105000E+01	.105000E+01	.805800E+02
13	2	0	.178000E+03	.139785E+03	.242000E+00	.105000E+01	.105000E+01	.785200E+02
14	2	0	.186000E+03	.104410E+03	.242000E+00	.105000E+01	.105000E+01	.795500E+02
15	2	0	.193000E+03	.819713E+02	.242000E+00	.105000E+01	.105000E+01	.805800E+02
16	2	0	.180000E+03	.118483E+03	.484000E+00	.157000E+01	.157000E+01	.785200E+02
17	2	0	.192000E+03	.885541E+02	.484000E+00	.156000E+01	.156000E+01	.800700E+02
18	2	0	.186000E+03	.101414E+03	.847000E+00	.311000E+01	.311000E+01	.785200E+02
19	2	0	.186000E+03	.100416E+03	.847000E+00	.311000E+01	.311000E+01	.785200E+02
20	2	0	.186000E+03	.984200E+02	.108500E+01	.311000E+01	.311000E+01	.785200E+02
21	2	0	.186000E+03	.834494E+02	.345000E+01	.311000E+01	.311000E+01	.785200E+02
22	2	0	.186000E+03	.804570E+02	.345000E+01	.311000E+01	.311000E+01	.785200E+02

TABLE A-VII

CORE SECTION GEOMETRICS

DATA FOR 20 HEAT CONDUCTING SLABS.

SLAB NUM	L VOL	R VOL	GEOM NUM / L C IND	STK IND / R C IND	LEFT SURFACE AREA, FT**2 / LFT HEATED EQ DIAMETER, FT	RIGHT SURFACE AREA, FT**2 / RHT HEATED EQ DIAMETER, FT	VOLUME FT**3 / LEFT CHANNEL LENGTH, FT	LEFT HYDRAULIC DIAMETER, FT / RIGHT CHANNEL LENGTH, FT	RIGHT HYDRAULIC DIAMETER, FT / INIT FRACT OF HEAT REMOVED	H.T. COEF BTU/H/F2/F	MAJOR L IN	MAJOR L OUT	JUNCTIONS R IN	JUNCTIONS R OUT
1	0	3	1	0	0.	.510000E+01	.180000E-01	0.	0.		0	0	2	1
			0	22	0.	0.	0.	0.						
2	0	4	1	1	0.	.640000E+01	.220000E-01	0.	0.		0	0	3	2
			0	22	0.	0.	0.	0.						
3	0	5	1	1	0.	.640000E+01	.220000E-01	0.	0.		0	0	4	3
			0	22	0.	0.	0.	0.						
4	0	6	1	1	0.	.510000E+01	.180000E-01	0.	0.		0	0	5	4
			0	22	0.	0.	0.	0.						
5	0	7	1	0	0.	.400000E+02	.135000E+00	0.	0.		0	0	7	6
			0	22	0.	0.	0.	0.						
6	0	8	1	1	0.	.400000E+02	.135000E+00	0.	0.		0	0	8	7
			0	22	0.	0.	0.	0.						
7	0	9	1	1	0.	.400000E+02	.135000E+00	0.	0.		0	0	9	8
			0	22	0.	0.	0.	0.						
8	0	10	1	0	0.	.320000E+02	.108000E+00	0.	0.		0	0	11	10
			0	22	0.	0.	0.	0.						
9	0	11	1	1	0.	.320000E+02	.108000E+00	0.	0.		0	0	12	11
			0	22	0.	0.	0.	0.						
10	0	12	1	1	0.	.320000E+02	.108000E+00	0.	0.		0	0	13	12
			0	22	0.	0.	0.	0.						
11	0	13	1	0	0.	.480000E+02	.162000E+00	0.	0.		0	0	15	14
			0	22	0.	0.	0.	0.						

TABLE A-VII (contd.)

CORE SECTION GEOMETRICS

12	0	14	1	1	0.	.480000E+02	.162000E+00	0.	0.	0	0	16	15
			0	22	0.	0.	0.	0.					
13	0	15	1	1	0.	.480000E+02	.162000E+00	0.	0.	0	0	17	16
			0	22	0.	0.	0.	0.					
14	0	16	1	0	0.	.920000E+02	.320000E+00	0.	0.	0	0	19	18
			0	22	0.	0.	0.	0.					
15	0	17	1	1	0.	.920000E+02	.320000E+00	0.	0.	0	0	20	19
			0	22	0.	0.	0.	0.					
16	0	18	1	0	0.	.161000E+03	.560000E+00	0.	0.	0	0	22	21
			0	22	0.	0.	0.	0.					
17	0	19	1	0	0.	.161000E+03	.560000E+00	0.	0.	0	0	24	23
			0	22	0.	0.	0.	0.					
18	0	20	1	0	0.	.200000E+03	.810000E+00	0.	0.	0	0	26	25
			0	22	0.	0.	0.	0.					
19	0	21	2	0	0.	.250000E+02	.133000E+02	0.	0.	0	0	28	27
			0	0	0.	0.	0.	0.					
20	0	22	2	0	0.	.750000E+02	.395000E+02	0.	0.	0	0	30	29
			0	0	0.	0.	0.	0.					

AXIAL STACKS OF HEAT SLABS =

1 THROUGH 4	1 DIMENSIONAL HEAT TRANSFER
5 THROUGH 7	1 DIMENSIONAL HEAT TRANSFER
8 THROUGH 10	1 DIMENSIONAL HEAT TRANSFER
11 THROUGH 13	1 DIMENSIONAL HEAT TRANSFER
14 THROUGH 15	1 DIMENSIONAL HEAT TRANSFER
16 THROUGH 16	1 DIMENSIONAL HEAT TRANSFER
17 THROUGH 17	1 DIMENSIONAL HEAT TRANSFER
18 THROUGH 18	1 DIMENSIONAL HEAT TRANSFER
19 THROUGH 19	1 DIMENSIONAL HEAT TRANSFER
20 THROUGH 20	1 DIMENSIONAL HEAT TRANSFER

TABLE A-VIII

JUNCTION DATA

JUNCTION DATA ACTUALLY BEING USED.

JUN NUM	FROM VOL	TO VOL	PUMP LEAK FILL	CHK VALV	INITIAL FLOW (LBM/SEC)	JUNCTION FLOW AREA (FT**2)	JUNCTION ELEVATION (FT)	JUNCTION DIAMETER (FT)	LEAK CONTRACTION COEFFICIENT
1	3	2	0	0	.841000E+02	.388000E-01	.785500E+02	.200000E-01	.100000E+01
2	4	3	0	0	.841000E+02	.388000E-01	.792900E+02	.200000E-01	.100000E+01
3	5	4	0	0	.841000E+02	.388000E-01	.801100E+02	.200000E-01	.100000E+01
4	6	5	0	0	.841000E+02	.388000E-01	.809400E+02	.200000E-01	.100000E+01
5	1	6	0	0	.841000E+02	.388000E-01	.816200E+02	.200000E-01	.100000E+01
6	7	2	0	0	.420500E+03	.194000E+00	.785500E+02	.200000E-01	.100000E+01
7	8	7	0	0	.420500E+03	.194000E+00	.795600E+02	.200000E-01	.100000E+01
8	9	8	0	0	.420500E+03	.194000E+00	.805900E+02	.200000E-01	.100000E+01
9	1	9	0	0	.420500E+03	.194000E+00	.816200E+02	.200000E-01	.100000E+01
10	10	2	0	0	.336400E+03	.155000E+00	.785500E+02	.200000E-01	.100000E+01
11	11	10	0	0	.336400E+03	.155000E+00	.795600E+02	.200000E-01	.100000E+01
12	12	11	0	0	.336400E+03	.155000E+00	.805900E+02	.200000E-01	.100000E+01
13	1	12	0	0	.336400E+03	.155000E+00	.816200E+02	.200000E-01	.100000E+01
14	13	2	0	0	.504600E+03	.233000E+00	.785500E+02	.200000E-01	.100000E+01
15	14	13	0	0	.504600E+03	.233000E+00	.795600E+02	.200000E-01	.100000E+01
16	15	14	0	0	.504600E+03	.233000E+00	.805900E+02	.200000E-01	.100000E+01
17	1	15	0	0	.504600E+03	.233000E+00	.816200E+02	.200000E-01	.100000E+01
18	16	2	0	0	.672800E+03	.310000E+00	.785500E+02	.200000E-01	.100000E+01
19	17	16	0	0	.672800E+03	.310000E+00	.800800E+02	.200000E-01	.100000E+01
20	1	17	0	0	.672800E+03	.310000E+00	.816200E+02	.200000E-01	.100000E+01
21	18	2	0	0	.588700E+03	.272000E+00	.785500E+02	.200000E-01	.100000E+01
22	1	18	0	0	.588700E+03	.272000E+00	.816200E+02	.200000E-01	.100000E+01
23	19	2	0	0	.586700E+03	.272000E+00	.785500E+02	.170000E-01	.100000E+01
24	1	19	0	0	.588700E+03	.272000E+00	.816200E+02	.170000E-01	.100000E+01
25	20	2	0	0	.761400E+03	.350000E+00	.785500E+02	.170000E-01	.100000E+01
26	1	20	0	0	.761400E+03	.350000E+00	.816200E+02	.170000E-01	.100000E+01
27	21	2	0	0	.743100E+03	.110400E+01	.785500E+02	.200000E-01	.100000E+01
28	1	21	0	0	.743100E+03	.110400E+01	.816200E+02	.200000E-01	.100000E+01
29	22	2	0	0	.730700E+03	.110400E+01	.785500E+02	.200000E-01	.100000E+01
30	1	22	0	0	.730700E+03	.110400E+01	.816200E+02	.200000E-01	.100000E+01

TABLE A-IX

JUNCTION DATA

JUNCTION DATA ACTUALLY BEING USED.

JUN NUM	VERT JUN INDX	CHOK -ING INDX	IC CALC INDX	MOM EQ. INDX	JUNCTION INERTIA	SP. ENERGY LOSS COEF (FORWARD)	SP.ENERGY LOSS COEF. (REVERSE)	RESIDUAL LOSS COEF. (NON-DIR)	RESIDUAL DELTA P (PSIA)	ENTHALPY TRANS INLET	ENTHALPY TRANS OUTLET
1	0	-1	3	0	.872813E+01	.998121E+00	.449577E+00	.795675E+00	.665696E+01	NO	NO
2	0	-1	3	0	.193631E+02	0.	0.	.233711E+00	.193896E+01	NO	NO
3	0	-1	3	0	.214555E+02	0.	0.	.191944E+00	.157999E+01	NO	NO
4	0	-1	3	0	.251422E+02	0.	0.	.103930E+00	.851757E+00	NO	NO
5	0	-1	3	0	.144990E+02	.449829E+00	.999242E+00	.640695E-01	.524753E+00	NO	NO
6	0	-1	3	0	.277638E+01	.990623E+00	.447885E+00	.805955E+00	.674003E+01	NO	NO
7	0	-1	3	0	.536720E+01	0.	0.	.287984E+00	.237755E+01	NO	NO
8	0	-1	3	0	.538720E+01	0.	0.	.189203E+00	.155164E+01	NO	NO
9	0	-1	3	0	.276820E+01	.449147E+00	.996215E+00	.342455E+00	.280483E+01	NO	NO
10	0	-1	3	0	.344346E+01	.992504E+00	.448310E+00	.803863E+00	.672936E+01	NO	NO
11	0	-1	3	0	.670135E+01	0.	0.	.290220E+00	.240085E+01	NO	NO
12	0	-1	3	0	.670135E+01	0.	0.	.189775E+00	.156003E+01	NO	NO
13	0	-1	3	0	.343528E+01	.449319E+00	.996975E+00	.338173E+00	.277692E+01	NO	NO
14	0	-1	3	0	.232159E+01	.988743E+00	.447450E+00	.813785E+00	.677579E+01	NO	NO
15	0	-1	3	0	.445762E+01	0.	0.	.296337E+00	.243960E+01	NO	NO
16	0	-1	3	0	.445762E+01	0.	0.	.194552E+00	.159216E+01	NO	NO
17	0	-1	3	0	.231341E+01	.448976E+00	.995454E+00	.348061E+00	.284586E+01	NO	NO
18	0	-1	3	0	.261099E+01	.985037E+00	.445621E+00	.898778E+00	.746259E+01	NO	NO
19	0	-1	3	0	.503642E+01	0.	0.	.462399E+00	.380660E+01	NO	NO
20	0	-1	3	0	.260281E+01	.448638E+00	.993954E+00	.293880E+00	.241320E+01	NO	NO
21	0	-1	3	3	.581699E+01	.986865E+00	.447035E+00	.160628E+00	.131967E+01	NO	NO
22	0	-1	3	0	.580881E+01	.448805E+00	.994695E+00	.544416E+00	.444586E+01	NO	NO
23	0	-1	3	0	.581699E+01	.986865E+00	.447035E+00	.945181E+00	.776306E+01	NO	NO
24	0	-1	3	0	.580881E+01	.448805E+00	.994695E+00	.329884E+00	.269393E+01	NO	NO
25	0	-1	3	0	.452135E+01	.983115E+00	.446185E+00	.934415E+00	.774903E+01	NO	NO
26	0	-1	3	0	.451317E+01	.448462E+00	.993176E+00	.307545E+00	.253734E+01	NO	NO
27	0	-1	3	0	.150809E+01	.947227E+00	.437965E+00	.215692E+02	.170581E+02	NO	NO
28	0	-1	3	0	.149991E+01	.445148E+00	.978554E+00	.289122E+02	.228356E+02	NO	NO
29	0	-1	3	0	.150809E+01	.947227E+00	.437965E+00	.223657E+02	.170697E+02	NO	NO
30	0	-1	3	0	.149991E+01	.445148E+00	.978554E+00	.299961E+02	.229076E+02	NO	NO

PARAMETERS IN JUNCTION MATRIX

NUMBER OF CHAINS	(MS) =	5
NUMBER OF CHAIN JUNCTIONS	(NTRI) =	11
NUMBER OF NON-CHAIN JUNCTIONS	(NQ) =	19
INDEX OF FIRST CRITICAL JUNCTION	(MPP) =	31
TOTAL NUMBER OF JUNCTIONS	(NTOT1) =	30

TABLE A-X

DATA VALUES FOR THREE HEAT CONDUCTOR GEOMETRIES

DATA FOR 3 HEAT SLAB GEOMETRIES

GEOM TYPE	REG NO	GAP IND	MAT NO	NO DX	XO TO N=1	REGION WIDTH	POWER FRAC
1	1		1	1	0.	.833000E-03	.978000E+00
	2	0	1	1		.125000E-02	.220000E-01
	3	0	2	1		.833000E-04	0.

SUM OF POWER FRACTIONS IS .100000E+01

GEOM TYPE	REG NO	GAP IND	MAT NO	NO DX	XO TO N=1	REGION WIDTH	POWER FRAC
1	1		1	1	0.	.750000E-02	.100000E+01

SUM OF POWER FRACTIONS IS .100000E+01

GEOM TYPE	REG NO	GAP IND	MAT NO	NO DX	XO TO N=1	REGION WIDTH	POWER FRAC
1	1		3	1	.100000E+00	.800000E+00	.100000E+01

SUM OF POWER FRACTIONS IS .100000E+01

TABLE A-XI

CORE GEOMETRY

RELAP4 Volume	Core Volume	Number of Fuel Elements	Volume Length (ft)	Fluid Volume (ft^3)	Flow Area (ft^2)	Flow Rate (lb/sec)	% of Total Reactor Power	Slab Metal Volume (ft^3)	Heat Transfer Area (ft^2)
3	1	1	0.67	0.026	0.0388	84.1	0.49	0.018	5.1
4	2	1	0.83	0.325	0.0388	84.1	0.97	0.022	6.4
5	3	1	0.83	0.325	0.0388	84.1	0.98	0.022	6.4
6	4	1	0.77	0.030	0.0388	84.1	0.61	0.018	5.1
7	5	5	1.05	0.202	0.194	420.5	4.46	0.135	40
8	6	5	1.05	0.202	0.194	420.5	5.95	0.135	40
9	7	5	1.05	0.202	0.194	420.5	4.46	0.135	40
10	8	4	1.05	0.161	0.155	336.4	3.42	0.108	32
11	9	4	1.05	0.161	0.155	336.4	4.56	0.108	32
12	10	4	1.05	0.161	0.155	336.4	3.42	0.108	32
13	11	6	1.05	0.242	0.233	504.6	4.84	0.162	48
14	12	6	1.05	0.242	0.233	504.6	6.45	0.162	48
15	13	6	1.05	0.242	0.233	504.6	4.84	0.162	48
16	14	8	1.57	0.484	0.310	672.8	8.90	0.320	92
17	15	8	1.57	0.484	0.310	672.8	8.90	0.320	92
18	16	7	3.15	0.847	0.272	588.7	11.40	0.560	161
19	17	7	3.15	0.847	0.272	588.7	7.92	0.560	161
20	18	14	3.15	0.085	0.350	761.4	13.44	0.810	200
21	19	R_I	Full	---	---	---	3.19	---	---
22	20	R_o	Full	---	---	---	0.80	---	---

TABLE A-XII

CORE POWER DISTRIBUTIONS

DATA FOR 20 CORE SECTIONS.

CORE SECT	SLAB NUM	PRINT T AT NODES			INITIAL CLAD THICKNESS	SECTION POWER FRACTION	PROMPT MOD FRACTION	DELAY MOD FRACTION
1	1	1	2	3	0.	.490000E-02	.300000E-01	.260000E+00
2	2	1	2	3	0.	.970000E-02	.300000E-01	.260000E+00
3	3	1	2	3	0.	.980000E-02	.300000E-01	.260000E+00
4	4	1	2	3	0.	.610000E-02	.300000E-01	.260000E+00
5	5	1	2	3	0.	.446000E-01	.300000E-01	.260000E+00
6	6	1	2	3	0.	.595000E-01	.300000E-01	.260000E+00
7	7	1	2	3	0.	.446000E-01	.300000E-01	.260000E+00
8	8	1	2	3	0.	.342000E-01	.300000E-01	.260000E+00
9	9	1	2	3	0.	.456000E-01	.300000E-01	.260000E+00
10	10	1	2	3	0.	.342000E-01	.300000E-01	.260000E+00
11	11	1	2	3	0.	.484000E-01	.300000E-01	.260000E+00
12	12	1	2	3	0.	.645000E-01	.300000E-01	.260000E+00
13	13	1	2	3	0.	.484000E-01	.300000E-01	.260000E+00
14	14	1	2	3	0.	.890000E-01	.300000E-01	.260000E+00
15	15	1	2	3	0.	.890000E-01	.300000E-01	.260000E+00
16	16	1	2	3	0.	.114000E+00	.300000E-01	.260000E+00
17	17	1	2	3	0.	.792000E-01	.300000E-01	.260000E+00
18	18	1	2	3	0.	.134400E+00	.300000E-01	.260000E+00
19	19	1	2	3	0.	.319000E-01	.300000E-01	.260000E+00
20	20	1	2	3	0.	.800000E-02	.300000E-01	.260000E+00

NO METAL WATER REACTION WILL BE CALCULATED

SUM OF POWER FRACTIONS = 1.0000000

The fraction of the prompt energy deposited directly in the moderator was input as 0.03. The fraction of the delayed energy deposited directly in the moderator was input as 0.26.

Metal-water reaction was not considered in the RELAP4 analysis. The fraction of the energy produced in the fuel plate cladding was input as 0.022. The oxide thickness was input as 1.0 mil. This oxide thickness represents a three-sigma standard deviation value for deposition on elements operated in the most recent ETR cycle.

7.4 Kinetics

The RELAP4 kinetics model has the capability for solving the kinetics equation described in the manual, incorporating possible sources of reactivity feedback. The code allows for input of any reactivity feedback source which may be significant.

The kinetics equation was solved for the ETR system using parameters defined in Table A-XIII. A β/ℓ value of 175 was used. The power calculation was based on one prompt neutron group, six groups of delayed neutrons, and eleven delayed gamma emitters. A value of 1.0 for the U-238 consumed per U-235 fission was used because the ETR fuel is highly enriched. The fraction of total power released at the time of fission was assumed to be 0.93 with 0.07 released as delayed gamma emitters.

Feedback due to void formation and feedback due to expansion of the fuel plates were considered. Water temperature feedback and the Doppler effect were considered insignificant.

To determine the reactivity effect of a reduction of moderator, void coefficients for each core volume were calculated from the core power distribution. These coefficients were used to determine the reactivity feedback from vapor voiding in the fuel element as well as feedback from displacement of the fuel plates due to increased temperature expansion. Because displacement is directly related to the fuel plate temperature, that feedback was input as reactivity per unit plate temperature change. Table A-XIV shows the calculated values of the coefficients for both voiding and displacement for each core volume and the method of calculation. The supplied void coefficients were based on data obtained at xenon equilibrium conditions when the core power distribution is flattest. Consequently the feedback will be conservative for a beginning of life power generation profile.

For calculation of the void feedback, the fraction of surface heat flux used to produce vapor bubbles was assumed to be 0.05, and the vapor bubble lifetime was assumed to be 0.001 second. These values were recommended for use in both highly subcooled and slightly subcooled flow in Reference A-6. The vertical profile used to determine void coefficients in the volumes distributed axially within each set of elements was the same as the profiles used for the heat generation values.

TABLE A-XIII

REACTOR KINETICS PARAMETERS

POWER TYPE	BETA OVER LIFETIME	INITIAL REACTIVITY	U-238 ATOMS CONSUMED PER U-235 ATOM FISSIONED
2	.175000E+03	0.	.100000E+01

FRACTION OF SURFACE HEAT FLUX UTILIZED IN PRODUCING SUBCOOLED VAPOR BUBBLES = 5.00000E-02

VAPOR BUBBLE LIFETIME (SEC) = 1.00000E-03

1 PROMPT AND 6 DELAYED NEUTRON GROUPS PLUS 11 DELAYED GAMMA EMITTERS
PROMPT ENERGY FRACTION = .930

REACTIVITY DATA TABLE 1

DATA PTS	TRIP ID	TIME	REACTIVITY	TIME	REACTIVITY	TIME	REACTIVITY
1	2	.100000E+04	0.				

REACTIVITY CURVE VERSUS NORMALIZED DENSITY.

DATA PTS.	NORMALIZED DENSITY	REACTIVITY (DOLLARS)	NORMALIZED DENSITY	REACTIVITY (DOLLARS)	NORMALIZED DENSITY	REACTIVITY (DOLLARS)
-2	0.	-.100000E+01	.100000E+01	0.		

DOPPLER REACTIVITY CURVE.

DATA PTS.	METAL TEMPERATURE	REACTIVITY (DOLLARS)	METAL TEMPERATURE	REACTIVITY (DOLLARS)	METAL TEMPERATURE	REACTIVITY (DOLLARS)
-2	.600000E+02	0.	.500000E+04	0.		

TABLE A-XIII (contd.)

REACTOR KINETICS PARAMETERS

CORE REG NUM	WATER DENSITY WEIGHTING FACTOR	DOPPLER WEIGHTING FACTOR	METAL TEMP. COEFFICIENT ($/DEG F)	WATER TEMP. COEFFICIENT ($/DEG F)
1	.813000E-01	0.	-.500000E-06	0.
2	.233000E+00	0.	-.140000E-05	0.
3	.232000E+00	0.	-.140000E-05	0.
4	.913000E-01	0.	-.560000E-06	0.
5	.819000E+00	0.	-.510000E-05	0.
6	.148600E+01	0.	-.920000E-05	0.
7	.807200E+00	0.	-.500000E-05	0.
8	.661000E+00	0.	-.410000E-05	0.
9	.119800E+01	0.	-.740000E-05	0.
10	.651000E+00	0.	-.400000E-05	0.
11	.949000E+00	0.	-.590000E-05	0.
12	.172100E+01	0.	-.106000E-04	0.
13	.935000E+00	0.	-.575000E-05	0.
14	.211000E+01	0.	-.130000E-04	0.
15	.209300E+01	0.	-.130000E-04	0.
16	.255700E+01	0.	-.158000E-04	0.
17	.149400E+01	0.	-.924000E-05	0.
18	.183600E+01	0.	-.114000E-04	0.
19	0.	0.	0.	0.
20	0.	0.	0.	0.

TABLE A-XIV

FEEDBACK PARAMETERS

RELAP4 Volume	Core Volume	Volume Fraction	$/Fraction of Void[a] at Xenon Equilibrium	Voiding ($/Fraction of Void) at Xenon Equilibrium	Displacement ($/°F) at Xenon Equilibrium
3	1	0.0046	17.666	0.0813	0.000109
4	2	0.0057	40.885	0.2330	0.000252
5	3	0.0057	40.703	0.2320	0.000251
6	4	0.0053	17.231	0.0913	0.000106
7	5	0.0355	23.083	0.8194	0.000143
8	6	0.0355	41.861	1.4860	0.000258
9	7	0.0355	22.738	0.8072	0.000140
10	8	0.0283	23.352	0.6610	0.000144
11	9	0.0283	42.348	1.1980	0.000261
12	10	0.0283	23.003	0.6510	0.000142
13	11	0.0426	22.278	0.9490	0.000138
14	12	0.0426	40.401	1.7210	0.000249
15	13	0.0426	21.945	0.9350	0.000135
16	14	0.0852	24.760	2.1100	0.000153
17	15	0.0852	24.521	2.0930	0.000152
18	16	0.1490	17.162	2.5570	0.000106
19	17	0.1490	10.025	1.4940	0.000062
20	18	0.1900	9.661	1.8360	0.000060

$$\text{Voiding} = (\text{Volume Fraction})\ (\$/\text{Fraction of Void})$$

$$\text{Displacement} = (\text{Fraction of Void}/°\text{F})\ (\$/\text{Fraction of Void})\ \left(\frac{0.050}{0.115}\right)$$

$$\text{Fraction of Void}/°\text{F} = \text{Coefficient of Thermal Expansion}$$

$$= 14.2 \times 10^{-6}/°\text{F}\ (68\text{–}572°\text{F})$$

$$\left(\frac{0.050}{0.115}\right) = \frac{\text{Width of the plate}}{\text{Width of the channel}}$$

[a] Void coefficients were based on data obtained at xenon equilibrium conditions when the core power distribution is the flattest.

7.5 Results

Due to instabilities in the RELAP4 code, the results could not be analytically determined for the complete loss of flow with no reactor scram. However, for purposes of subsequent calculations, 50% of the core was assumed to melt.

For the case with loss of flow and reactor scram, no core melting is expected. The loss of flow incident does not postulate a break in the primary containment. Thus, fission products released to the primary coolant are assumed to leak to the building commensurate with the maximum allowable system leakage of 75 gpm (a conservative estimate).

With 50% of the core melting after complete loss of coolant flow, the amount of fluid leakage to the containment building is determined as follows:

7.5.1 ETR Loss of Flow.

(1) Assumptions.

(a) Reactor does not scram on low vessel delta pressure

(b) Reactor does not scram from high pressure.

(c) Reactor does not scram from high surge tank level.

(2) Estimated Sequence of Events.

Time (sec)	Event
0	Complete loss of flow, reactor operation for 30 days.
25	Flow has coasted down to zero. Fuel plate temperature in hot volume = 425°F. Reactor power = 44 MW.
26	Fuel plate insulates due to steam formation.
27	Fuel plate reaches melting temperature at hot spot.
30	Estimated 50% of core has reached melting temperature. Estimated reactor power is 20 MW.

(3) Assumption. 5°F rise in ETR system temperature results in a 25-psig rise in pressure. System volume is about 55,000 gallons or 455,000 pounds.

Coefficient of thermal expansion, $\beta = 0.418 \times 10^{-3}$ per degree F.

Rate of temperature rise = $150^{o}F/hr$.

Rate of volumetric rise = 461 ft^3/hr; this is 3,448.5 gal/hr or 57.5 gpm.

Therefore, liquid release rate to the containment = 57.5 gpm, which is well within the permissible leakage limit of 75 gpm.

Since the reactor can operate with 75 gpm leakage, the assumption can be made that all fluid is leaked to the containment at a rate of 75 gpm, with a source equivalent to 50% core melting.

8. CONSEQUENCES

Although 50% of the ETR core has been assumed to melt, the fission product release from the primary containment is rather tenuous because of the tortuous path the fission products must take before release occurs. With the integrity of the primary containment remaining intact except for the ruptured relief valves, fission products are entrained in the primary coolant which has been assumed to leak into the reactor pipe tunnel at a rate of 75 gpm. Since the reactor coolant system (RCS) has been depressurized, the leak continues as water (instead of steam) at the 75 gpm rate at which water is being added to the core.

The solids portion of the fission-product inventory will remain in the water, thus posing no potential for offsite exposure. The noble fission gas inventory, contained in 50% of the core, is pessimistically assumed to be contained in the first 4,500 gallons which will leak during the first hour. Actually, more realistic behavior of the noble fission gases would be to evolve from the water within the primary containment system and thus largely be contained within this containment. Of the 50% core halogen inventory, half of that present is assumed to evolve from the primary coolant as it is released from the RCS. For all the fission products released from the RCS, credit for no holdup or scavenging is assumed for the pipe tunnel or heat exchanger building. Thus, the leak rate of the released fission products to the atmosphere is 0.0139%/sec for the noble fission gases and 0.069%/sec for the halogens and is assumed to continue for a period of one hour. Under the most probable circumstances, the fission products would be exhausted out of the pipe tunnel to the heat exchanger building, then out the stack.

Once the material has been released from the stack top, resulting radiological doses to offsite individuals are determined probabilistically. The location of an individual is a factor in the determination of his radiological dose with his distance and direction from the release point being important considerations. The meteorological weather regime (inversion or lapse) and the wind speed that persists at that time of release are also factors that determine the radiological dose.

The nearest site boundary (where an individual may be located potentially) is approximately 1.12×10^4 meters distant from the Test Reactor Area (TRA) at a direction west-northwest of TRA. The nearest population center to TRA is Arco, Idaho, which is 13^o north of west and lies about 2.74×10^4 meters distant. Probability of lapse and inversion weather conditions existing at TRA, based on an annual basis, are 52.1 and 47.9%, respectively. With the wind speed residing in the denominator of the diffusion equation, the resulting radiological doses are inversely proportional to the wind speed. Figures A-10 and A-11 show how diffusion calculations depend on both the direction and speed of the wind. Although these wind roses are based on data taken at Central Facilities Area (CFA), they are applicable to TRA, which is only 4.1 miles distant from CFA. The directions indicated on the wind roses are the directions from which the wind is blowing, that is, a north wind is blowing from north to south.

In summary, the consequence probabilities are multiplied together to obtain the overall probability that an individual at some location will receive a radiological exposure. This probability has been obtained for illustration purposes for an individual at Arco, Idaho.

Probability for lapse condition	=	0.52
Probability for wind direction	=	0.0125
Probability for 6 m/sec wind speed	=	0.5
Overall probability	=	0.00325

The radiological consequences for this downwind position (at Arco) are summarized in Tables A-XV through A-XVIII. The calculated thyroid inhalation dose is 2.13 rem and the cloud γ whole body dose is 9.3×10^{-3} rem. The cloud γ doses reported here are those calculated by the semi-infinite cloud model.

These results were obtained with use of RSAC-2[A-7] a computer program capable of determining radiological hazards resulting from a reactor system accident. The program generates a fission product inventory from a given set of reactor operating conditions, and then computes the various gamma dosages and ingestion parameters specific to the facility location and environmental condition.

9. RISK RESULTS

Risk, the chance of injury, has been defined mathematically as follows:

Risk = (accident probability)(accident consequence).

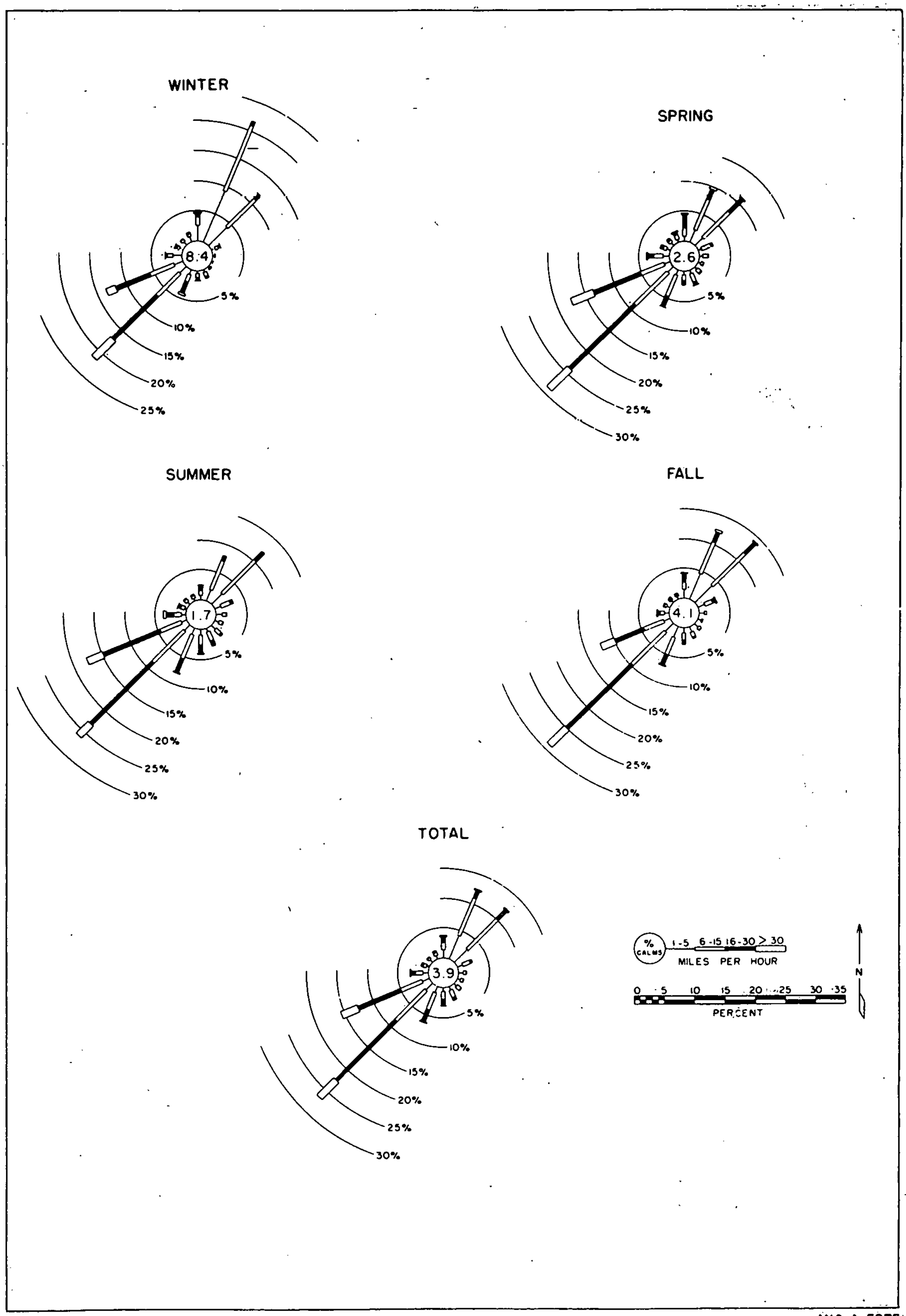

Fig. A-10 CFA 250-foot level lapse wind roses, July 1951 through May 1962.

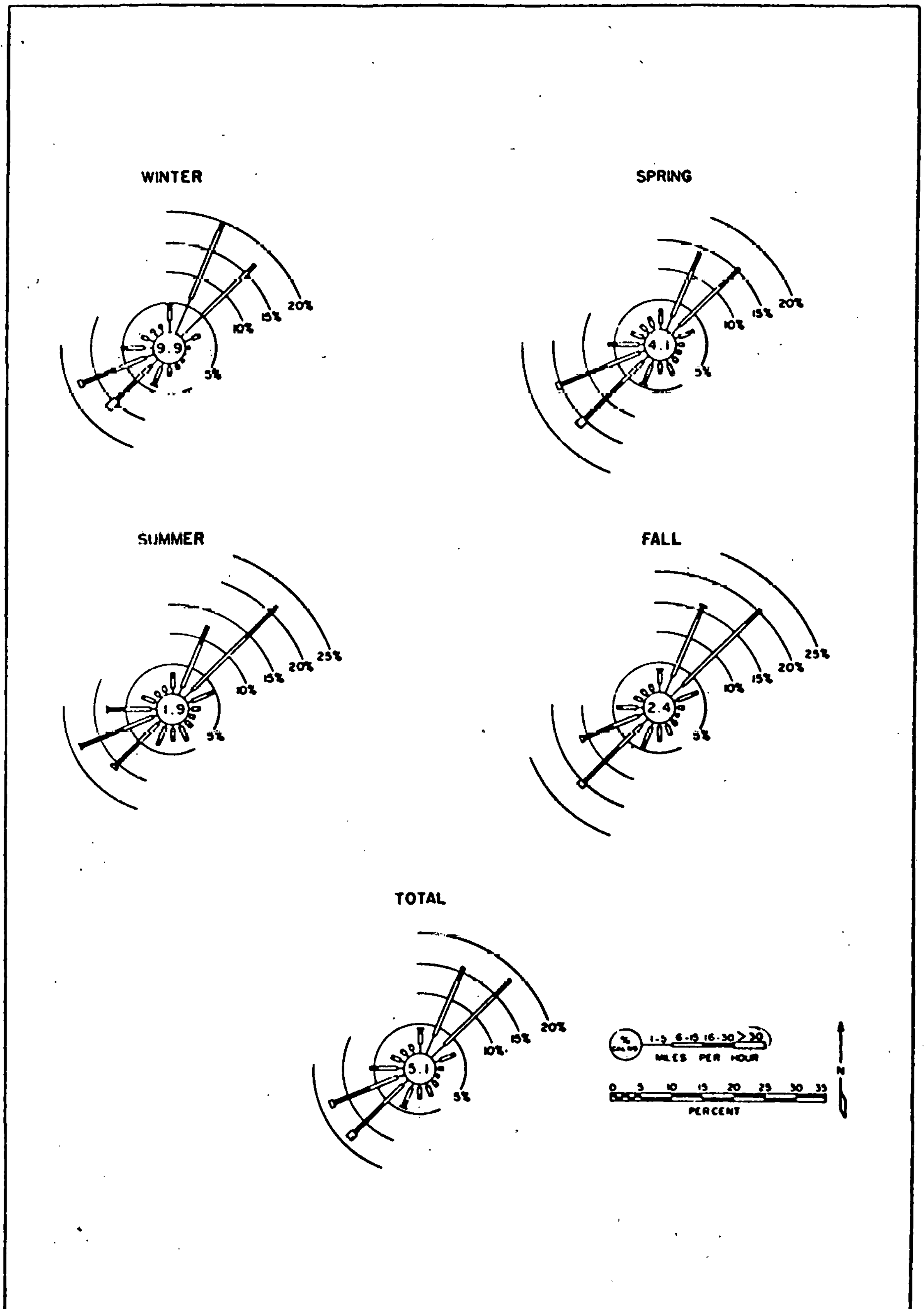

Fig. A-11 CFA 250-foot level inversion wind roses, July 1951 through May 1962.

ANC-A-5876

TABLE A-XV

DATA INPUT TO RSAC-2

```
(RSAC-2/03/26/74/D.R.WENZEL)       RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM (RSAC-2)            DATE 06/20/75
ETR RISK ANALYSIS ( LAPSE WEATHER CONDITION )                                                        PAGE    1
    CHAIN DATA INPUT IS FROM DATA SET     55555
***CURIE CALCULATION
    THE REACTOR HAS OPERATED AT  1.750E 08 WATTS FOR  2.592E 06 SECONDS
         WATTS TO FISSIONS/SECOND CONVERSION FACTOR =  3.120E 10
    REACTOR HAS BEEN SHUT DOWN FOR  1.000E 00 SECONDS
    FRACTIONATION BY ELEMENT GROUP -- PERCENT
         SOLIDS =  0.0              HALOGENS =  2.500E 01     NOBLE GASES =  5.000E 01     CESIUM =  0.0
    TOTAL FISSION PRODUCT RELEASE =  2.360E 18 D/SEC OR  6.380E 07 CURIES
***METEOROLOGICAL DATA
    UBAR =  6.000E 00 (M/SEC)      W1 =  1.500E 00      W2 =  1.500E 00      W3 =  1.500E 00
    STACK HEIGHT =  7.600E 01 (M)      INVERSION HEIGHT =  0.0          (M)      WASHOUT FACTOR =  0.0
    DOWNWIND DISTANCES -- METERS
      C.0              1.120E 04       2.740E 04
     THERE IS 1 SET OF EXPONENTIALS (K, P)
            2.778E-04         0.0
    PASQUILL CLASS B METEOROLOGY, MARKEE SIGY(S), MARKEE SIGZ(S)
    BUILDING WIDTH =  1.000E 00 (M)   BUILDING HEIGHT =  1.000E 00 (M)   DILUTION FACTOR =  1.000E 00
         SIGY AND SIGZ(S) -- METERS
      1.519E 03        2.651E 03       3.362E 03       5.851E 03
         CY AND CZ(S)
      1.973E 00        3.443E 00       2.232E 00       3.886E 00
***DECAY CALCULATION -- CURIES
CHAIN  1   MASS NUMBER   3
      NUCLIDE         PERCENT FISSION YIELD      DECAY CONST (SEC-1)         HALF LIFE     GAMMA ENERGY (MEV/DIS)
      1   H-  3               8.800E-03                1.791E-09           1.226E 01 YRS             0.0
             TIME (SEC)            H-  3
               0.0                3.01E 01
               1.867E 03          3.01E 01
               4.567E 03          3.01E 01
CHAIN  2   MASS NUMBER  72
      NUCLIDE         PERCENT FISSION YIELD      DECAY CONST (SEC-1)         HALF LIFE     GAMMA ENERGY (MEV/DIS)
      1   ZN- 72              1.600E-05                4.140E-06           4.650E 01 HRS           9.100E-01
      2   GA- 72              0.0                      1.365E-05           1.410E 01 HRS           2.700E 00
```

TABLE A-XVI

THYROID INHALATION DOSE WITH

LAPSE WEATHER CONDITIONS

(RSAC-2/03/26/74/D.R.WENZEL) RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM (RSAC-2) DATE 06/20/75

ETR RISK ANALYSIS (LAPSE WEATHER CONDITION) PAGE 21

NUCLIDE	PERCENT FISSION YIELD	DECAY CONST (SEC-1)	HALF LIFE	GAMMA ENERGY (MEV/DIS)
1 EU-159	1.050E-03	6.417E-04	1.800E 01 MIN	0.0
2 GD-159	0.0	1.069E-05	1.800E 01 HRS	4.600E-01

TIME (SEC)	EU-159	GD-159
0.0	0.0	0.0
1.867E 03	0.0	0.0
4.567E 03	0.0	0.0

TOTAL FPI VS TIME (TIME, FPI)
0.0 6.38E 07 1.867E 03 2.94E 07 4.557E 03 2.53E 07

TOTAL FPI BY GROUP -- CURIES

TIME (SEC)	SOLIDS	HALOGENS	NOBLE GASES	CESIUM
0.0	0.0	1.94E 07	4.44E 07	0.0
1.87E 03	1.95E 06	9.51E 06	1.69E 07	
1.04E 06	4.57E 03	2.04E 06	8.00E 06	
1.47E 07	5.86E 05			

SUMMATION OF FPI BY ELEMENT
SUMMATION OF ISOTOPES C (TIME, FPI)
1.469E 07 1.87E 03
SUMMATION OF ISOTOPES*** (TIME, FPI)
7.805E 06

***INHALATION CALCULATION (ICRP II MODEL FOR SOLUBLE NUCLIDES)

BREATHING RATE = 3.470E-04 (CU M/SEC) INHALATION TIME = 3.600E 03 (SEC) MULTIPLIER = 1.000E 00

DOWNWIND DISTANCE = 1.120E 04 (M) TIME = 1.867E 03 (SEC) CHI/Q = 1.317E-08 (SEC/CU M)

THYROID DOSE FOR EACH I ISOTOPE (ATOMIC NO 53)

ISOTOPE	CHAIN	DOSE (REM)
I - 129	60	3.68E-09
I - 129	61	2.89E-08
I - 131	63	5.47E 00
I - 131	64	9.56E-01
I - 132	65	6.18E-02
I - 133	66	2.20E 00
I - 133	67	1.35E 00
I - 134	68	1.42E-02
I - 135	69	1.40E-01
I - 135	70	3.77E-01

THYROID INHALATION DOSE FOR ALL I ISOTOPES (ATOMIC NO 53) IS 1.0574E 01 (REM)

TOTAL THYROID INHALATION DOSE = 1.0574E 01 (REM)

DOWNWIND DISTANCE = 2.740E 04 (M) TIME = 4.567E 03 (SEC) CHI/Q = 2.697E-09 (SEC/CU M)

TABLE A-XVI (contd.)

THYROID INHALATION DOSE WITH LAPSE WEATHER CONDITIONS

(RSAC-2/03/26/74/D.R.WENZEL) RADIOLOGICAL SAFETY ANALYSIS CCMPUTER PROGRAM (RSAC-2) DATE 06/20/75

ETR RISK ANALYSIS (LAPSE WEATHER CONDITION) PAGE 22

THYROID DOSE FOR EACH I ISOTOPE (ATOMIC NO 53)

ISOTOPE	CHAIN	DCSE (REM)
I - 129	60	7.95E-10
I - 129	61	5.92E-09
I - 131	63	1.12E 00
I - 131	64	1.95E-01
I - 132	65	1.01E-02
I - 133	66	4.40E-01
I - 133	67	2.70E-01
I - 134	68	1.60E-03
I - 135	69	2.65E-02
I - 135	70	7.15E-02

THYROID INHALATICN DOSE FOR ALL I ISOTOPES (ATOMIC NO 53) IS 2.1321E 00 (REM)

TOTAL THYROID INHALATION DOSE = 2.1321E 00 (REM)

***GAMMA DCSE CALCULATION WITH EXPOSURE TIME = 3.6000E 03 (SEC)

DOWNWIND DISTANCE = 1.120E 04 (M) DOSE = 9.11E-02 (REM)

DCWNWINC CISTANCE = 2.740E 04 (M) DOSE = 3.33E 01 (REM)

***GAMMA DOSE CALCULATION WITH EXPCSURE TIME = 3.6000E 03 (SEC)
CALCULATIONS MADE USING ONLY THE SEMI-INFINITE MODEL

DOWNWIND DISTANCE = 1.120E 04 (M) DCSE = 5.85E-02 (REM)

DCWNWINC CISTANCE = 2.740E 04 (M) DOSE = 9.32E-03 (REM)

EXECUTION TIME = 1.717414E 02

TABLE A-XVII

INPUT DATA TO RSAC-2

```
(RSAC-2/03/26/74/D.R.WENZEL)       RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM (RSAC-2)        DATE 06/19/75
ETR RISK ANALYSIS (INVERSION WEATHER CONDITION)                                                  PAGE    1
   CHAIN DATA INPUT IS FROM DATA SET    55555
***CURIE CALCULATION

   THE REACTOR HAS OPERATED AT  1.750E 08 WATTS FOR  2.592E 06 SECONDS
       WATTS TO FISSIONS/SECOND CONVERSION FACTOR =  3.120E 10

   REACTOR HAS BEEN SHUT DOWN FOR  1.000E 00 SECONDS

   FRACTIONATION BY ELEMENT GROUP -- PERCENT
       SOLIDS =  0.0              HALOGENS =  2.500E 01      NOBLE GASES =  5.000E 01     CESIUM =  0.0

   TOTAL FISSION PRODUCT RELEASE =  2.360E 18 D/SEC OR  6.380E 07 CURIES
***METEOROLOGICAL DATA

   UBAR =  2.000E 00 (M/SEC)     W1 =  1.500E 00     W2 =  1.500E 00     W3 =  1.500E 00
   STACK HEIGHT =  7.600E 01 (M)    INVERSION HEIGHT =  3.000E 02 (M)     WASHOUT FACTOR =  0.0

   DOWNWIND DISTANCES -- METERS

     0.0              1.120E 04       2.740E 04

    THERE IS 1 SET OF EXPONENTIALS (K, P)
        2.778E-04        0.0

   PASQUILL CLASS H METEOROLOGY, MARKEE SIGY(S), MARKEE SIGZ(S)

   BUILDING WIDTH =  1.000E 00 (M)   BUILDING HEIGHT =  1.000E 00 (M)   DILUTION FACTOR =  1.000E 00
       SIGY AND SIGZ(S) -- METERS
     1.273E 03        1.409E 01       2.352E 03        1.850E 01
       CY AND CZ(S)
     1.654E 00        1.832E-02       1.562E 00        1.229E-02
***DECAY CALCULATION -- CURIES

CHAIN  1   MASS NUMBER   3
     NUCLIDE          PERCENT FISSION YIELD      DECAY CONST (SEC-1)        HALF LIFE     GAMMA ENERGY (MEV/DIS)
     1    H-  3               8.800E-03                1.791E-09          1.226E 01 YRS            0.0

              TIME (SEC)              H-  3
                0.0                 3.01E 01
                5.600E 03           3.01E 01
                1.370E 04           3.01E 01

CHAIN  2   MASS NUMBER   72
     NUCLIDE          PERCENT FISSION YIELD      DECAY CONST (SEC-1)        HALF LIFE     GAMMA ENERGY (MEV/DIS)
     1   ZN- 72               1.600E-05                4.140E-06          4.650E 01 HRS          9.100E-01
     2   GA- 72               0.0                      1.365E-05          1.410E 01 HRS          2.700E 00
```

TABLE A-XVIII

THYROID INHALATION DOSE WITH INVERSION WEATHER CONDITIONS

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(RSAC-2/03/26/74/D.R.WENZEL)      RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM (RSAC-2)          DATE 06/19/75
ETR RISK ANALYSIS (INVERSION WEATHER CONDITION)                                                     PAGE  2
    NUCLIDE          PERCENT FISSION YIELC       CECAY CONST (SEC-1)          HALF LIFE    GAMMA ENERGY (MEV/DIS)
    1  EU-159               1.C5CE-03                  6.417E-04           1.800E 01 MIN           0.0
    2  GD-159               0.0                        1.069E-05           1.800E C1 HRS           4.600E-01

           TIME (SEC)           EU-159            GD-159
            0.0                 0.0               C.0
            5.600E 03           0.0               0.0
            1.37CE C4           0.0               0.0

     TCTAL FPI VS TIME (TIME, FPI)
     C.C          6.38E 07     5.600E 03      2.41E 07      1.370E 04      1.88E 07

     TOTAL FPI BY GROUP -- CURIES
         TIME (SEC)        SOLIDS           HALOGENS         NOBLE GASES        CESIUM
       C.C                 0.0               1.94E 07          4.44E C7          0.0
        5.6CE C3           1.95E 06          7.58E 06          1.42E 07
        4.26E C5           1.37E C4          1.15E 06          5.70E 06
        1.19E 07           2.52E 04

     SUMMATION OF FPI BY ELEMENT
         SUMMATION OF    ISOTOPES  C (TIME, FPI)
           1.469E 07        5.60E 03
         SUMMATICN CF    ISCTCPES*** (TIME, FPI)
           5.633E 06
***INHALATICN CALCULATION (ICRP II MODEL FCR SCLUBLE NUCLIDES)
      BREATHING RATE =  3.470E-04 (CU M/SEC)     INHALATION TIME =  3.600E 03 (SEC)     MULTIPLIER =  1.000E 00
      DOWNWIND DISTANCE =  1.120E 04 (M)         TIME =  5.600E 03 (SEC)        CHI/Q =  4.372E-12 (SEC/CU M)
                                          THYROID  DOSE FOR EACH  I ISCTOPE (ATOMIC NO 53)
                                           ISOTOPE      CHAIN        DCSE (REM)
                                          I - 129        60          1.29E-12
                                          I - 129        61          9.59E-12
                                          I - 131        63          1.81E-03
                                          I - 131        64          3.16E-04
                                          I - 132        65          1.50E-05
                                          I - 133        66          7.06E-04
                                          I - 133        67          4.33E-04
                                          I - 134        68          2.06E-06
                                          I - 135        69          4.16E-05
                                          I - 135        70          1.12E-04
              THYROID  INHALATICN DOSE FOR ALL  I ISOTOPES (ATOMIC NO 53) IS  3.4370E-03 (REM)
         TCTAL THYROID  INHALATION DOSE =  3.4370E-03 (REM)
      DCWNWIND CISTANCE =  2.740E 04 (M)         TIME =  1.370E 04 (SEC)        CHI/Q =  7.974E-10 (SEC/CU M)
```

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TABLE A-XVIII (contd.)

THYROID INHALATION DOSE WITH INVERVSION WEATHER CONDITIONS

(RSAC-2/03/26/74/C.R.WENZEL) RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM (RSAC-2) DATE 06/19/75

ETR RISK ANALYSIS (INVERSION WEATHER CONDITION) PAGE 3

THYROID DOSE FOR EACH I ISOTOPE (ATOMIC NO 53)

ISOTOPE	CHAIN	DOSE (REM)
I - 129	60	2.35E-10
I - 129	61	1.75E-09
I - 131	63	3.28E-01
I - 131	64	5.72E-02
I - 132	65	1.38E-03
I - 133	66	1.19E-01
I - 133	67	7.33E-02
I - 134	68	6.29E-05
I - 135	69	6.02E-03
I - 135	70	1.63E-02

THYROID INHALATION DOSE FOR ALL I ISOTOPES (ATOMIC NO 53) IS 6.0127E-01 (REM)

TOTAL THYROID INHALATION DOSE = 6.0127E-01 (REM)

***GAMMA DOSE CALCULATION WITH EXPOSURE TIME = 3.6000E 03 (SEC)

DOWNWIND DISTANCE = 1.120E 04 (M) DOSE = 2.47E 00 (REM)

DOWNWIND DISTANCE = 2.740E 04 (M) DOSE = 8.79E-01 (REM)

EXECUTION TIME = 6.744191E 01

Risk is, therefore, the probability of the accident occurring, times the probability of the released material reaching a given population, times the doses in rems that will reach the given population.

The information needed for this calculation has been given earlier in the report and is collected here for an example risk calculation. The probability of the accident occurring for the three event sequences was presented in Section 6. The consequences are discussed in Section 8.

The event tree earlier shown in Figure A-4 is repeated in Figure A-12 with tabulated values for the risk calculation. The risk for Sequences 1 and 2 (S_1 and S_2) is 0 because a radioactivity release will not occur. A risk for S_3 has been calculated for the nearest population zone which is Arco, Idaho, at a distance of 2.74×10^4 meters from the ETR. The release for the lapse weather condition, 2.13 rems (from Table A-XVI), is used in the calculation. The hazard probability of 3.25×10^{-3}, tabulated in Section 8, denotes the probability of the released radioactivity reaching Arco.

The resultant probability of a radioactive exposure of 2.13 rem to the populace of Arco, Idaho, resulting from a loss of flow accident at the ETR during lapse weather conditions is 3.25×10^{-10}/operating cycle. When the exposure and the probability are multiplied together, the risk becomes 6.9×10^{-10}.

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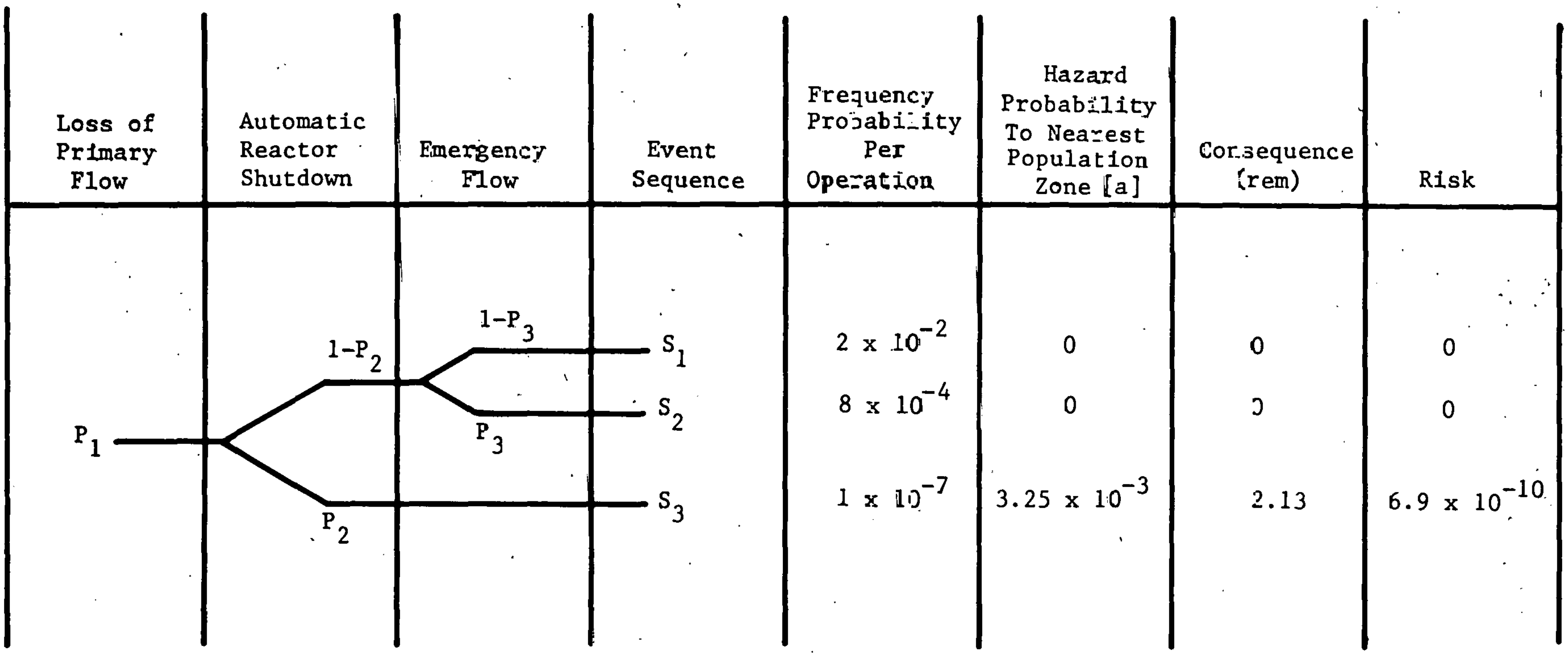

[a] Arco, Idaho 2.74×10^4 meters (17 miles) ANC-A-5877

Fig. A-12 Risk summary for ETR loss of flow event.

APPENDIX B

DIFFERENCES BETWEEN TEST REACTORS AND COMMERCIAL NUCLEAR PLANTS

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APPENDIX B

DIFFERENCES BETWEEN TEST REACTORS AND COMMERCIAL NUCLEAR PLANTS

During the course of this study a few differences were observed between test reactors and the large commercial nuclear power plants pertinent to risk assessment that are worthy of mention. These important differences are summarized as follows:

(1) Because of the relatively long operating cycle (dictated by refueling requirements), the commercial power plant designs provide for on-line testing of most components associated with engineered safety features. Test reactors, for the most part, do not have provisions for on-line tests. Instead, most test and maintenance activities are performed for the test reactors before each experiment (while the reactor is shut down). Thus, test outages can influence system unavailabilities for the commercial power plant; they are not so significant with test reactors.

(2) Test reactors are subject to more frequent startups and shutdowns; therefore, components characterized by cyclic failure rates are more likely to fail in a given time frame.

(3) Test reactor operations are less routine due to the variety of experimental test programs encountered, thus more operational attention is required in the control and surveillance of the plant.

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